Another area where we're concerned about is more of a traditional I will call it grid convergence or dischordization convergence issue in the degree of spatial variation or spatial dischordization and temporal dischordization.

In spatial dischordization, it's not classical like you might do in solving a PDE, but look at the number of infiltration bins they've constructed or the number of thermal hydrology bins. It's how they've chopped and diced up their system. If you increase the number of bins, does it have an effect on your dose versus time curve? That's the sort of information we need to see to be confident.

Likewise with time stepping, I gave you a couple of examples here, just the time step size for seepage corrosion, spent fuel. I probably would eliminate the thing about climate states. That's more or less fixed by what the climate has done in the past.

Let me go on to the next slide and talk about --

CHAIRMAN HORNBERGER: Wait, wait, wait.

MR. WITTMEYER: Yes.

CHAIRMAN HORNBERGER: Before you go on.

MR. WITTMEYER: Yes.

CHAIRMAN HORNBERGER: I'm just curious. Your first point there is stability of the first moment in terms of realizations. Is that because you don't care about any higher moments? Because we know it's going to be more difficult to get stability in the higher moments.

MR. WITTMEYER: Well, the regulation, the proposed regulation is on the first moment, and I think as we've discussed a number of other times, there's some interesting things about that first moment though. At early times it's a very high percentile value, oftentimes exceeding 95 percent. Later times it drops down a little bit lower.

But that is the objective, is to look at the first moment and the stability of that first moment.

I think I already expressed some of the concerns here on convergence, statistical convergence of the LHS method. I think we would just like to see a little greater rigor applied to demonstrating convergence. One example that I might give is instead of doing a single 100 realization run and then comparing it to a 500 or a 1,000 is to do a series of ten 100 realization runs and then a series of ten 500 realization runs and show me how the variance decreases as a function of the number of realizations.

That's, I think, a more typical thing you might expect to see.

In terms of submodels, the ones that were used to compute biosphere dose conversion factors, and you might want to reverse the C and the D there on that BCDF, and the sat zone transport kernels, they have a limited number of those things. The sat zone transport kernels are computed off line, then abstracted from a fairly complex code. I don't know what the number is. I think it's around 100.

If you had 1,000, would it make a significant difference? You need to know whether that's been looked at.

One thing that I have here is oftentimes when you're dealing with limited sample sizes the mean will be affected by extreme values, and the more you do up to a limit, the more samples you observe, the more likely you are to see an extreme condition. I'm not suggesting that's an unbounded problem. It should be bounded, but when you're talking about finite sample sizes, you can see a significant amount of variation.

So we really do need to see or DOE needs to demonstrate the peak mean dose does not increase significantly as the spatial resolution is increased, and likewise with the temporal resolution, which can be more of a standard convergence test that a modeler might do to make sure he's got the right grid resolution.

We discussed these issues with DOE at the last technical exchange held in the first week of August, and DOE has planned to describe the types of statistical measures they will use in the method that they're going to employ when they demonstrate statistical convergence of their LHS approach, and they're going to try and conduct the appropriate analyses and provide some additional documentation that show that the PA code or their TSPA code is stable as you change the number of bins or other things where you bid, chop up, dice, et cetera, parts of the TSPA code.

MR. FIRTH: Okay. Sitakanta.

MR. MOHANTY: My name is Sitakanta Mohanty. I will be talking to you about model validation and computer code verification as a part of overall performance objective subissue.

At the beginning, let me mention that this effort was not just pursued by the TSPA group, but rather, we got information from various KTIs, including the TSPA KTI.

The proposed Part 63 of the regulation does not explicitly specify requirements for code verification or model validation. However, it is disguised in several forms in the text, and that calls for testing and corroboration of confidence in performance assessment model, and that DOE should provide technical basis for the performance assessment models. That means model support.

And there should be confirmation that barrier behavior is consistent with assumptions made in the performance assessment approach and model.

I would be a little bit careful in talking about validation and verification because the issue of validation has been pursued and debated quite a bit in the industry for many, many years. Therefore, I'm going to switch from validation verification to model confidence or software confidence.

We did get some criticism at the last technical exchange. So just to be on the side of the scientist right now, we would rather talk about model confidence, but what I'm talking about is model validation or software verification.

From a regulatory point of view, we would like to make a clear distinction between scientific validation and what the regulation expects. So, therefore, we would like to highlight that in the context of regulation, a scientific model validation is not sought after, but rather it can be a part of it, but it's not all of it.

We also noticed that there was quite a bit of confusion about the definition because there are scientific definitions for software verification and model validation as realized quality assurance definitions. There have been terms used such as software validation, software verification, model validation, model verification. So we had a little bit of difficulties at the beginning to communicate with the DOE. So, therefore, we set some working definitions so that we can communicate the findings from NRC side to DOE.

So these are the two working definitions that we use. Building software confidence or software verification is a process that provides assurance that the computer code is performing correctly the operations that were specified in the numerical model.

And a parallel definition for building confidence in the model is that it is a process that provides assurance that the model as embodied in the computer code that is verified is the correct representation of the process or the system for which it is intended.

So the model can be both a conceptual model, as well as the mathematical model.

DR. GARRICK: This strikes me as a very sensible way to approach it given the failures that have existed in trying to come up with a prescriptive process for validating models, codes, and what have you, but I wanted to ask. Maybe you answered this. I'm a little show in keeping up with you.

The codes that DOE uses in their TSPA, have they been verified in the traditional code verification sense that is customary? I'm used to the utility world where there is a very specific QA requirement for verification of software codes. Do these codes have that kind of documentation at least?

MR. MOHANTY: I think my short answer would be yes, but we do have several findings that what I'm going to go over --

DR. GARRICK: Okay.

MR. MOHANTY: -- one by one.

MR. FIRTH: I guess I would also add that a number of the software codes that are feeding the TSPA, DOE has a corrective action report in terms of the validation, which is the procedures that they have set up to qualify in terms of making sure that they have confidence in the software that they're using.

And the corrective action report also calls into question the possibility that TSPA-SR may have a similar condition. So DOE is working on that through their quality assurance process.

DR. GARRICK: thank you.

MR. MOHANTY: In terms of guidance, there is no technical position yet from NRC's point of view on model validation, which perhaps is a larger topic than software verification.

However, there is a NUREG 1636 that was related. It was a joint product of staff members from the NRC and from the SKI. That's a white paper that has example approaches or strategy for model validation. However, that is not intended to be viewed as a staff position because it does have one example so that perhaps some can view as perhaps the only approach, but that's not the only approach to model validation.

There were several findings. However, these are some of the summaries of the findings. The general NRC NCWA (phonetic) staff believe was that there is element of software, confidence building software verification approach in the DOE's TSP-SR, but the staff believe is maybe there is trying to have a more rigorous verification that is yet to be accomplished.

Also, the verification appears to be limited to only a component model when you look at it from a TSPA model point of view, and the verification appears to have been done only for selected models or the abstracted models in the TSPA model.

And also there were limited sets of hand calculations. Perhaps these are hand calculation sets where picked rather randomly, depending on which model was -- which model implementation was verified.

As far as the model confidence or model validation is concerned, their model validation was primarily documented in the analysis model reports. So that would refer to validation at the process model level, perhaps to some extent to the abstraction.

Also, it was documented rather in an abbreviated form in the science and engineering report. DOE does have quality assurance requirements which is documented in the AP3.10Q, but a parallel document did not exist for the software verification.

The only point I would like to mention here is that NRC' review of the AP3.10Q suggests that the peer review is considered as an equal alternative to experiment or field studies.

And these are two of the corrective actions that Jim just mentioned. One is the failure to consistently implement the quality assurance procedure. That means they do have a procedure, but it's a matter of how uniformly that procedure was implemented by the staff in developing their TSP-SR document.

And also there was failure to implement QA program related to software, and there was lack of effective independent validation and verification. These are the findings of DOE's QA staff.

Here is a list of NRC's concerns. Perhaps the most over arching concern that NRC conveyed at the technical exchange is that DOE does not appear to have or perhaps DOE has not conveyed a particular vision to the technical staff as to what is to be accomplished under model validation and software verification.

And because of the lack of this philosophy or a strategy, perhaps there is a huge variation in the implementation of that in the document. And DOE did indicate that they do have procedures, but procedures do not always convey a vision or philosophy for staff members to adopt.

And as far as the software confidence

building is concerned, it's the same thing that there was no plan for the software verification that NRC's aware of, and there were errors found in the hand calculation. These were conveyed to DOE even prior to the technical exchange.

The first one is for model abstraction. Ranges were used. Those were outside the range or outside the values for which the models were developed. Although there were explicit statements in the document that abstracted models are used within the range that were developed, but that was not always the case.

And another significant finding was that significance of the warning or errors in the DOE TSPA software, which in the GoldSim was not explained or why there were errors in the error log.

Therefore, it was difficult to trace and understand the origin of those errors, and the consequence of those errors as far as the results were concerned.

And TSPA model correctness was not verified for the extremes. By that what I mean is that most of the verifications were performed by using the mean value data set, but in NRC's own experience, most of their -- a lot of errors are found, if you exercise the code in the sensitivity analysis or uncertainty analysis mode because it takes the code to some regions in the data space that one would not go otherwise.

And there were no --

CHAIRMAN HORNBERGER: Sitakanta, I guess I'm a little unclear as to what you mean by verified in that last comment. I mean, what you're really talking about is, I assume, getting a warm and fuzzy feeling that the model doesn't do strange things.

MR. MOHANTY: I think the most important part of verification is to find bugs in the code.

CHAIRMAN HORNBERGER: Okay.

MR. MOHANTY: Just the simplest way to explain it.

CHAIRMAN HORNBERGER: Yeah, but I meant but still what you're talking about there, I think, is that you're not satisfied that DOE did enough. You can never find all the bugs in a complex code. We know that.

MR. MOHANTY: Yes, yes.

CHAIRMAN HORNBERGER: But you're saying that they didn't do enough to make sure that there weren't some very obvious bugs; is that --

MR. MOHANTY: Right.

MR. ESH: They didn't write it down. Maybe they did at the time.

MR. MOHANTY: And also we observed that there were no verification at the TSPA code level. That means most verifications were confined to individual abstracted models.

But if you take bigger chunks of the code, they might stumble into other bugs in the code.

Here are some of the NRC concerns as far as the model confidence or model validation is concerned. One of the concerns was that perhaps a rigorous model confidence building effort was not initiated or taken up by the DOE.

And DOE model validation requirements, although specified in their QA procedure, AP3.10Q, but those were not consistently implemented for all models or for the TSP model.

Another concern NRC had and conveyed to DOE was that the way the documents were written, that implied that as long as you are validating your abstracted model, the TSPA model is automatically validated, something NRC decided to be done. That's because a sum of the subsets doesn't make a complete set when the TPA code.

And also the conceptual models were

validated. For some of these abstracted models, all the process models, but the corresponding mathematical models were not validated, and biosphere is one example. That is not the case for all models, but in some cases, that is something we observed.

And there was no objective comparisons for component model, say, with data or with code comparison with other codes or calibration, that some of the aspects that were represented at validation can very well fit under calibration, perhaps not under validation.

And peer review appeared to have been used, as I mentioned already, as a substitute for objective information that is reasonably available, such as experimental data or natural analog studies.

CHAIRMAN HORNBERGER: Can you say just a little more about that? Does that mean that some of their people said, "Yeah, that's about right," instead of doing comparisons with data?

MR. MOHANTY: Well --

CHAIRMAN HORNBERGER: Or does it mean a formal peer review?

MR. MOHANTY: Well, it can be both ways. It can be internal peer review or external peer review because in all -- if there are high level of uncertainty in data, if the data collection can be so expensive that it cannot be justified, I think the AP3.10Q procedure specifies that they will take the peer review approach.

CHAIRMAN HORNBERGER: Right, but I mean that's sort of an expert elicitation, and NRC has a formal procedure that they accept. You're not negating that; is that correct?

MR. MOHANTY: Right. I'm not quite sure. Maybe somebody else can answer this better. I don't know. I think internal peer review can also be used, but I don't know if NRC's regulation applies to that internal peer review procedure. But every time there has been an external peer review, the issue of following the guidance has been brought up.

CHAIRMAN HORNBERGER: Okay.

MR. FIRTH: Well, I guess I would add that peer review is also different than expert elicitation in terms of this context.

CHAIRMAN HORNBERGER: Yes.

MR. FIRTH: So in order for -- and DOE has within their QA procedures peer review as an option, and our concern is that in some cases that they're using that sort of as a preferred way when we would be like -- we would like to see objective evidence that the code is producing results that you would expect to see.

CHAIRMAN HORNBERGER: Yeah, I agree. I'm just curious because if anyone does a serious peer review, it's hard to imagine -- for me to imagine -- that those peer reviewers are not going to be aware that there is objective information that is reasonably available and call the whole thing into question.

I mean, I can't imagine a serious peer review that would rubber stamp something if there was objective information that hadn't been brought to the table.

MR. FIRTH: Mike, can you?

MR. LEE: Yeah. This is Mike Lee.

I was one of the co-authors of the white paper. What we're saying in the whit paper basically is DOE is to use a sliding scale in determining what it needs to validate. For things that are most important to performance, we think there is need for more confidence building rather than less.

In the context of what constitutes confidence building, the white paper lays out some approaches to acquiring that confidence, be it experimental data, natural analog studies, field tests, acquisition of more data.

Last, but not least, we do acknowledge that they can conduct peer review, but that's if there's no other opportunities or avenues to collect information to provide that confidence.

MR. MOHANTY: Finally, at the technical exchange several resolutions or agreements were reached, both under software confidence, building a software verification and model validation.

Here are some of the agreements that were reached. Under software confidence building, DOE agreed to provide an overall strategy or plan for the PA software verification approach.

There was some issue with regard to how they are going to document this. In the final agreement, there was no specificity mentioned as to which document will have that plan.

And we also agreed to provide minimum requirements for completing software verification. That means DOE has to sort of put a stamp indicating, okay, we have done the -- we have completed the verification because we have reached these goals that were identified in the plan of similar documents.

And also DOE will describe how software verification actions and results will be documented. This issue was brought up because TSP-SR is already out, but DOE was asked a question as to whether the TSPA-SR document is going to be revised to build in this new software confidence building effort, and the answer was no.

There may be some other documents that will come up which I don't think we got much specificity about those.

Similarly, for model confidence building effort, DOE has agreed to provide a plan and justification for the selection of validation criteria, both qualitative and quantitative, or whether any validation is needed for a particular model or not, those would be documented in the plan.

And DOE will provide evidence that the validation approach is consistent with the quality assurance requirements stipulated in their AP3.10Q.

And also, once it is implemented, then DOE agreed that it will document somewhere indicating that these are the criteria, and these are the validation results, and therefore, validation results meet validation criteria.

That's all I have to present.

DR. GARRICK: Thank you now.

We're a few minutes beyond our schedule, and therefore, we're not going to have hopefully too many questions at this time. I know we all have questions.

I have one, for example, on this business of confidence building and whether or not you've given any thought to what would be a reasonable metric for that. If we can build risk models and probabilistic models for things as complex as repositories and nuclear power plants and space shuttles, chances are we could probably come up with some sort of a more quantitative manifestation of confidence in these models than just a warm, fuzzy feeling.

But let's not get into that now. That's just something that I think, like software reliability is an evolving science, and there's some interesting techniques out there for addressing it, and I'm curious as to whether or not any of them would apply here.

Are there any burning questions on the part of committee members, staff, or the audience?

(No response.)

DR. GARRICK: What I would like to say is as you can tell form the questions, the committee is very interested in this. This is an exercise that we consider very important, and we're anxious to hear from you again when there is a little bit of insight into how the resolution process really proceeds and how convergence is taking place because when you see a situation that's generating several hundred agreements in a relatively short period of time, you can't help but wonder what kind of a situation we're really getting into and how this is going to play out in terms of the convergence of those agreements.

So we appreciate, Jim, what you and your team have done here today. It was an excellent insight into the whole process. Most of the committee members, in fact, all of the committee members have observed the technical exchange process, including the consultant, and it's something that's very high on our list as an indication of how things are proceeding. And so we're anxious to follow it pretty closely.

So we thank you for spending the time, and we hope to see all of you again.

MR. FIRTH: Okay. Thank you for the opportunity.

DR. GARRICK: George.

CHAIRMAN HORNBERGER: Thanks, John.

Yeah, I had lots f questions, but John cowed me into not asking them.

(Laughter.)

CHAIRMAN HORNBERGER: Actually we do have to take a break. We have several things that we need to get done.

I would just like to add my thanks to all of you. Excellent presentations, and as John said, we really are interested in this, and as Dave said, we recognize that we can only get ice cubes here, but we do look at the agreements, and if we have detailed questions, we will get back to you.

Thanks very much.

We're going to now adjourn until two o'clock.

(Whereupon, at 12:36 p.m., the meeting was recessed for lunch, to reconvene at 2:00 p.m., the same day.)

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

(2:00 p.m.)

CHAIRMAN HORNBERGER: The meeting will come to order for the afternoon session.

Our first topic, as Howard Larson announced this morning, we have revised our agenda. So we are going to launch right into a discussion on preclosure plans, and our cognizant for our favorite topic of preclosure is Milt Levenson. I will turn it over to Milt.

DR. LEVENSON: Well, I really don't have any introductory remarks to make. We'll save the nasty questions for later. So go ahead.

MR. GUNTER: Good afternoon. My name is Tim Gunter. I'm with the Department of Energy in Las Vegas. I'm with the Office of Licensing and Regulatory Compliance, and I'm on the Regulatory Interactions and Policy Development Team.

Since this is my first time addressing the committee, I thought I would just take a real brief 30 seconds and tell you a little bit about myself.

I've been with the Department of Energy at Las Vegas for three years. One year before that I was there as on-site representative for DOE Environmental Management, and prior to that I was at Savannah River site where I was in engineering related to the Defense Waste Processing Facility, the vitrification facility for high level waste.

Prior to my DOE experience, I had 12 years of work at Charleston Naval Shipyard, civil service there in the Nuclear Engineering Department, where I spent seven years in test engineering, and then the five years following that, I was a branch supervisor for radiological performance assessment, which is not the same kind of performance assessments as we've been talking about here, but it's more auditing type function.

And then educational background is I have a degree in nuclear engineering from Georgia Tech.

CHAIRMAN HORNBERGER: You must have graduated when you were 14 with all of those years.

(Laughter.)

MR. GUNTER: Well, it does seem like a lot of years, and sometimes I feel like time is going by to fast.

With that, I'd like to go ahead and get into our presentation. This is actually a two-part presentation. In my role now with DOE, I'm basically the lead for interactions with the NRC. So I coordinate, you know, most of the KTI meetings now.

I worked with Carol Hanlon earlier when she was doing that, and now she's primarily the lead for SR, and I've taken the KTI lead and other NRC interactions.

The second part of the presentation is going to be given by Paul Harrington, who is the DOE technical lead for this area.

Okay. On page 2 of the presentation is just an overview of what I'm going to talk about: a little bit of introduction; talk about the objectives of the technical exchange for the preclosure issues; the structure, how the NRC set up the preclosure safety area; what we did to prepare to get ready for the technical exchange; talk a little bit about the items that we discussed; the results of the meeting; and then the agreements that NRC and DOE came to.

Page 3 is the introduction and objectives. This technical exchange, as you know, is one in a series of prelicensing meetings that are related to NRC's issue resolution and sufficiency review. This preclosure meeting was held July 24th to 26th this year, and this was the first technical exchange related to preclosure where we actually came to agreements.

We've had several previous meetings on these topics, either technical exchanges or more informal Appendix 7 type of meetings where we discussed information and our approaches and plans for some of the topics, but there were no agreements reached, nor was there any intent to reach agreements at that time.

So the objectives, the last two bullets there, basically the objectives, we went into this wanting to reach agreement with the NRC on the structure, the preclosure area, and then to discuss a limited number of topics that they had indicated to us that they wanted to discuss and also respond to a series of questions that they provided to us in advance.

Page 4 continues with the structure of the preclosure safety area. To track the status of agreements, NRC has broken the preclosure safety area into ten topics, and they identify to us in a letter from Bill Reamer to our Assistant Manager, Steve Brocoum, which was dated April 27th, 2001.

One of the reasons they did this is because there's no existing issue resolution status report to date that's been issued for this area. It's not considered a key technical issue, but it is an areas of importance to the NRC and us.

The NRC further stated that we could get additional information. They would further define the subtopics and acceptance criteria that should be addressed on a potential license application, and they would do this through future meetings and also in the Yucca Mountain review plan once that's issued.

And as I mentioned, the April letter identified the subtopics to be discussed during the first meeting.

On page 5 is a list of the topics that NRC put in their letter that they consider preclosure safety topics. The ones that have an asterisk by them are the ones that they indicated they would like to discuss at this first meeting.

Real briefly, they include the site description, and most of these come or seem to lie generally with proposed Part 63, the sections on safety analysis reports and ISA.

So other topics they wanted to hear about were the description of the SSCs and operational process activities, how we identify hazards and initiating events, the identification of the event sequences, how we approach our consequence analysis of these, and then some information on identification of this as season as far as the safety, safety controls and measures to insure availability of safety systems, and also the design of those SSCs.

Page 6 is a continuation. This is the last three topics. These were not discussed at this technical exchange, but they indicated these would be topics for future meetings, and that's meeting the 10 CFR, Part 20, as low as reasonably achievable requirement for normal operations in Cat. 1 event sequences, our plans for retrieval and alternate storage of radioactive waste, if necessary, and eventually the plans for permanent closure and decontamination and dismantlement of the surface facilities.

As far as preparations for the technical exchange on page 7, the NRC staff provided us several pieces of information. They provided talking points that they would present in the meeting. They provided to us a number of preliminary comments and questions that their staff had, and they would like us to address.

And then they provided some draft staff positions on issues, which included a risk significance categorization and a paper on level of design detail.

In response to that, DOE staff prepared preliminary responses to the NRC questions, and then we discussed those at the meeting.

Page 8 is a list of the specific topics that we discussed or if you want to call them subtopics, they're topics for the area. But anyway, they included the integrated safety analysis process. This was a general overview presentation that we did just to sort of set the stage and provide general information as to our identification of hazards, analysis of frequency and consequences, and then how that feeds into the design criteria.

More specifically, getting into the area that the NRC has specific questions on and included the aircraft crash hazards. They had questions on the types of aircraft that were of potential concern, and they had questions on projections in the future flight activities and how we would address that, and other activities, such as combat training and things that potentially could go on in the immediate area.

The other topics we discussed included tornado missiles. NRC had questions on our bases for the design basis event. They had questions on our categorization and our use of the wind speed and frequencies for the tornado we selected.

We also talked about how we screen out events by design and justification of probability estimates. Both of these fall under the identification of event sequence topic.

We had discussions on consequence analysis, you know, how we calculate doses under the Cat. 1 and Cat. 2 DBEs.

And then the last two topics on the identification of the SSCs important to safety and those important to waste isolation.

DR. LEVENSON: Question.

MR. GUNTER: Yes.

DR. LEVENSON: What components are important to safety as different from waste isolation? What safety issues are we talking about here?

MR. GUNTER: Well, part of the safety, as defined by Part 63, is, you know, those SSCs that help you meet or make sure you don't exceed the dose limits for the Cat. 1.

DR. LEVENSON: But how is that separate from the waste isolation?

MR. GUNTER: Well, the waste isolation is those components that, you know, retard. There's different limits in Part 63 for like long-term dose and the dose to the public versus operating doses, you know, that would be important to safety.

MR. RICHARDSON: I can give some examples there if it would help.

MR. GUNTER: Okay.

MR. RICHARDSON: One example would be obviously the drip shield, which is presently credited in the TSPA analysis, and therefore, it would be identified as important to waste isolation.

DR. LEVENSON: Right.

MR. RICHARDSON: However, obviously that doesn't at present come into play during the preclosure period and would not be an SSC identified as important to safety from the preclosure safety viewpoint.

So there are different SSC that my be identified in post closure that aren't identified in preclosure. Obviously there's some that are common, like the waste package, which has various safety functions credited both in preclosure safety and obviously in post closure. So that's an example of an SSC that would, I guess, be identified both under important to safety and also important to waste isolation, and its various safety functions would have to be combined then for its total safety basis.

DR. LEVENSON: Okay. As you're defined it here, important to safety is limited to preclosure safety.

MR. RICHARDSON: Yes, as defined in the proposed regulation, yes.

CHAIRMAN HORNBERGER: Could you give us an example of something that would be identified or might be identified as important in preclosure to waste isolation, but not important to safety?

MR. RICHARDSON: Oh, certainly. The waste handling building is probably one of the better examples.

CHAIRMAN HORNBERGER: It's important to waste isolation, but not --

MR. RICHARDSON: I'm sorry. It's important to safety, but not waste isolation. Is that what you asked?

CHAIRMAN HORNBERGER: No, I asked the converse.

MR. RICHARDSON: Well, that's the example I gave, the drip shield is important to waste --

CHAIRMAN HORNBERGER: Preclosure?

MR. RICHARDSON: It's important to waste isolation, but not preclosure.

CHAIRMAN HORNBERGER: Oh, waste isolation is not preclosure.

MR. RICHARDSON: Right.

CHAIRMAN HORNBERGER: Now I'm with you.

MR. GUNTER: Important to safety is preclosure --

CHAIRMAN HORNBERGER: I thought this was a preclosure discussion.

MR. RICHARDSON: It is.

PARTICIPANT: It's horrible language. That's what it is.

CHAIRMAN HORNBERGER: I thought we were only discussing preclosure. That's why I thought --

MR. RICHARDSON: Well, the reason this comes up is during the technical exchange they wanted to know how we would do identification both for important -- for all SSCs whether it comes up in preclosure or post closure, and it's handled under two different topics there.

DR. LEVENSON: Well, I guess that contributes to the confusion because if you're going to talk about waste isolation, that includes almost everything in all of the other tech. exchanges. So how do you -- where's the boundary between preclosure and everything else?

PARTICIPANT: Waste isolation is the repository.

DR. LEVENSON: Yeah, that's all of the KTIs and all of the tech. exchanges.

MR. GUNTER: The NRC in their letter provided the specific areas they wanted to discuss and the specific topics they wanted to discuss in that meeting. So that's what we tried to address.

DR. GARRICK: What they're saying is, "Don't blame us. Blame the NRC."

DR. LEVENSON: Well, but you know, items important to waste isolation, that's the geology and that's everything. Did you really attempt to respond to that?

MR. GUNTER: No, we didn't attempt to respond to everything, and Paul Harrington would like to add to this.

MR. HARRINGTON: Yeah, this is Paul Harrington, DOE.

That was Dennis Richardson, the contractor preclosure safety manager who spoke earlier.

We do have to address not only items important to safety, but also items important to waste isolation during the preclosure period. I think NRC properly put that down as a topic that we'll have to cover. We didn't do it at this meeting, as Tim said, but we need to insure during the preclosure period that we're not going to do something to those systems, structures, and components and even the natural features that are important to waste isolation, to degrade their ability to perform their action during the post closure period.

Fabrication of a waste package is obviously important to waste isolation component, but during the preclosure period is where we fabricate it, do all of the testing and placement handling and everything else.

So I think it's entirely appropriate. I'm not going to blame the NRC that we have the ITWI issue as part of a preclosure discussion at some point. The important safety items though are really limited to the preclosure components that don't play a role in post closure.

DR. LEVENSON: Well, I can see a basis for a great deal of confusion because you're now going to have a significant number of items that are reviewed in two different tech. exchanges. I don't know how you keep things from getting muddled, and you're not the only place that the waste container is going to be discussed, et cetera. You have different groups of people discussing the same issue at different meetings. It seems to me that's --

MR. GUNTER: That's true not only for this, but a lot of other areas where there's a lot of overlap, you know, just by the nature of the issue. It affects a lot of different KTIs.

DR. LEVENSON: Okay.

MR. HARRINGTON: For waste package, for example, in the other meetings I expect the discussion would really focus on how can we assure ourselves that the waste package performs as expected during that extended post closure period.

For the preclosure discussion of the waste package, I would expect a discussion to go toward how do we insure fabrication of it so that we can get the performance that we're expecting to get. How do we handle it and that sort of thing?

We haven'[t seen the details of those sets of questions yet though. So we'll deal with that when we get them.

MR. GUNTER: Any other questions at this point?

(No response.)

MR. GUNTER: Okay. I think I was ready to move to slide nine. This is the additional items that were discussed. The differentiated approach to providing information in the license application, this is an approach to where we would provide a different degree of details, one at the construction authorization and then as compared to the application to receive and possess. We would expect at CA we would not be providing the extent or the level of detail for a lot of the information that we would later on.

The next topic was preclosure criticality and burn-up credit. One of the big issues here which I'll discuss a little bit later, but had to do with an open item in NRC's SER on our topical report for the disposal criticality analysis methodology, and that is the need to perform surveys or measurements of all the fuel assemblies to take any credit for burn-up.

Other areas that they wanted to discuss was flooding, the probabilities, the technical basis for beyond design basis events.

And then the final topic which we discussed a little bit already, but waste package, design and fabrication of the waste package. This included things like the drop test, fabrication methods, welding methods, NDEs, and in the post weld treatments, weld repairs.

The results of the meeting on page 10. Calling the discussions at this technical exchange, NRC stated that it agreed with DOE's general integrated safety analysis methodology for several items that are listed there.

One, the identification of event sequences; the use of the mean value for probability distributions for categorization; our methodology for consequence analysis; and our methodology for identification and the categorization of systems in regards to the quality levels; and the categorization of the SSCs or the safety in regards to quality levels.

They also stated that they were in general agreement with our proposed differentiated approach where the information that we would provide in the license application for a construction authorization and that for an application to receive and possess nuclear waste, and then also the level of design detail for the SSCs to be included in the license application.

Page 12 is what we considered a couple of issues. We say issues in that we may not be 100 percent in agreement between NRC staff and DOE staff at this time. We still have some work to do in this area or these areas, and the first one has to do with the burn-up credit, and that is although the NRC staff stated that they agree that the utility records are more accurate and reliable indicator of commercial spent nuclear fuel burn-up, the current NRC position is that physical measurements would be required to verify the reactor records for fuel assemblies for those that we want to claim burn-up credit.

And they've also indicated that this should be a 100 percent verification. This comes out of NUREG 3.71, where they included this requirement that is in addition to what they implement or refer to in that NUREG to ANSI ANS standards, 8.17.

DOE believes that 100 percent measurements are not necessary based on the fact that we are in agreement with NRC that the reactor records are the most accurate method of determining this.

As I mentioned earlier this is an open item in the NRC's SER, and we have agreed to come up with a proposal as to how we would address this in the future and provide this to NRC. This would be our approach to how we would, you know, have confirmation of the burn-up.

We don't know what we propose at this time, but it would be something other than 100 percent measurements.

DR. LEVENSON: That's the concept for this physical measurement? How would this be done? A constructive method or --

MR. GUNTER: That's one of the things that we have to come u with. We don't have -- currently we don't have a method to do that. So it's part of our approach in coming up with the method to measure it. We have to come up with the physical means to measurement.

DR. LEVENSON: What's the purpose of the measurement?

MR. GUNTER: The purpose is to confirm that the fuel assemblies, you know, actually have been burned to the level that the reactor records indicate, and we take credit for this burn-up, you know, in our analyses for criticality controls.

DR. LEVENSON: So it's a criticality control issue?

MR. GUNTER: Right. But NRC has stated they're willing to --

DR. LEVENSON: Was there any discussion whether a measurement of -- a simple measurement of K effective would be acceptable, which might be significantly easier than trying to do burn-up?

If it's for criticality, that's why I asked what's the purpose of it.

MR. GUNTER: Yeah, I don't remember a discussion of, you know, measurement of K effective per se. I mean the approach --

DR. LEVENSON: It would be a much more accurate measurement if there's a criticality potential or not.

MR. GUNTER: Well, you would have to measure -- somehow you would have to measure, you know, what the inventory of isotopes and your fissile materials are. You would still -- there would still be some calculations involved to determine, you know, the K effective.

DR. GARRICK: Are you suggesting that there will be a different procedure for handling the spent fuel depending on whether it's low burn-up or high burn-up?

MR. GUNTER: I don't think the procedure is any different.

DR. GARRICK: I can't imagine -- from a criticality standpoint, that's a quantitative effect, not a qualitative effect. I can't imagine you'd handle the fuel any less careful, for example, a low burn-up fuel versus a high or vice versa.

MR. GUNTER: No, it wouldn't change the way we handle the fuel or the procedures. Where it comes into play is in our analyses for, you know, doing criticality and the consequences of criticality, you know.

DR. LEVENSON: Is this another case where you make some measurements to put into the record if it makes no difference on what you do and how you handle the fuel?

MR. GUNTER: It's a confirmatory measurement that the fuel assemblies have been, in fact, burned. They're not somehow a new fuel assembly.

DR. LEVENSON: Well, whether it's a new fuel assembly or been burned, you have a very simple measurement by the gross radiation field. So there's no question that you're going to know if you've got an unburned subassembly in there without any additional measurements.

MR. GUNTER: Well, right. It would be fairly simple if it was just like a go-no go kind of measurement where you could determine whether it had been burned or had not been burned, but our understanding is they want, you know, a more detailed measurement, one that actually confirms, you know, the burn-up as compared to the calculations by the reactor records.

CHAIRMAN HORNBERGER: I suppose the question though is suppose you come up with a measurement scheme and suppose you make a measurement that fails to confirm that this particular assembly has not been burned to the extent that you think. Does it make any difference at all, any difference at all to how you handle that assembly?

MR. GUNTER: I don't know if we know the answer to that. I mean, I would say no, but --

DR. LEVENSON: Let me put some numbers on.

MR. GUNTER: Okay.

DR. LEVENSON: Suppose the paper records said it's 50,000 megawatt days per ton burn-up, and it turns out it's only 25,000 megawatt days per ton. Does that makes any difference at all?

MR. GUNTER: I expect what we would have to do is, you know, run those numbers back through our criticality calculations. I wouldn't expect particularly one assembly would make any difference. If you had a number of assemblies, they potentially could.

Again, this is, in my view, just a confirmatory measurement that they're asking for.

DR. LEVENSON: Well, it's one thing to confirm something on which you're going to act, and something else to confirm something which doesn't make any difference.

DR. GARRICK: So really what you're saying is the reason for it is to be able to justify your computations, your calculation.

MR. GUNTER: Right.

DR. GARRICK: But it has nothing to do with the way in which you handle the stuff.

MR. GUNTER: Right. I can't foresee of a way it would impact the way we handle the fuel. It's more the analyses.

DR. GARRICK: Well, I would hope not.

MR. HARRINGTON: We recognize the potential for getting different values from the reactor records versus some measurement that would be done. That was one of the issues about if you do the measurement and you believe that it's probably a less accurate indicator than what you pull out of the records, what do you do then when you get a delta?

We haven't yet some to, I think, even a decision within our own house, let alone resolution with the NRC staff on what to do in that indication or in that case. We understand that it's a very real possibility, but I don't have an answer for you.

MR. RAHIMI: This is Meraj Rahimi, NRC, in the area of criticality.

The gross gamma type of a measurement,

that's not acceptable to us because the DOE is designing the waste package internal basket components as a function of the burn-up. So they have to know the exact burn-up in order to see where the assembly is placed on the loading curve.

So they need to determine or verify the reactor records with respect to the burn-up.

In terms of if they do verification measurement and it turns out for the three and a half percent enriched fuel that they are loading they need at least 40 gigawatt days burn-up and it turns out to be like 25, and they have to place that assembly in their other basket design, which is, say, let's say, designed for the fresh field assumption, which is in essence a 21 PWR assembly. It holds a 12 PWR assembly.

Unless they can investigate, pinpoint, you know, what the problem is, why the reactor record shows, you know, 40 gigawatt days and, you know, their measurement shows 25, and they can pinpoint it to, you know, a source term or whatever, but other than that, if they can't do anything about it, they have to place it in a waste package that has not been designed to take credit for 40 gigawatt days, three and a half percent enriched fuel.

DR. LEVENSON: What's the allowable K effective in the waste package from staff position?

MR. REAMER: The allowable K effective as DOE has proposed for the preclosure part is .95 K effective, subtracting all of the biases from it, the code biases. So, you know, doing all of the validation and determining all of the code biases for the 21 PWR waste package, it might be like .92.

DR. LEVENSON: And how much does the burn-up have to be off to go from there to critical? A pretty big damn number, isn't it?

MR. REAMER: Yeah. I mean, to do the -- I mean, if you do the same analysis, you know, assuming those 21 PWR assemblies, I mean, you have to do the calculations. Certainly if they were fresh, it would be critical above one.

DR. LEVENSON: Yeah, but it's very easy to confirm they're not fresh.

MR. REAMER: Right.

DR. LEVENSON: That's easy. I mean -

MR. REAMER: Yeah, with the gross gamma, right. The go-no go, but in terms of the way that the DOE has selected to do is a function of the enrichment. They have a loading curve. They say if the assembly is three and a half percent enriched, they have designed a basket that for the assembly to have at least 35 gigawatt days burn-up. I mean, they go based on their loading curve.

I mean, their design, it is as a function of the burn-up and enrichment.

DR. LEVENSON: You're saying that this didn't originate with the staff. You're just -- this is what DOE is proposing, and you're saying in order to meeting it they have to do this.

MR. REAMER: That's right. Yeah, in order, you know, they are making the waste package very efficient in order to have a high payload waste package. So they are -- I mean, they could have done a sort of fresh fuel assumption.

DR. LEVENSON: Is there any feeling about how accurate might be the age of the fuel compared to its burn-up? Would you expect that the age might be more reliability from the records, how long since it came out of a reactor?

MR. REAMER: Well, the age, I mean, DOE from what we understand is not going to take credit as long as the fuel, you know, is five years old or older. They are not going to take credit for the reactivity decrease as a function of age.

DR. LEVENSON: Oh, I wasn't thinking that.

I was just thinking if you knew the age reasonably reliably, then a gross gamma measurement might be a reasonable way to estimate burn-up within the accuracy that you need.

MR. REAMER: Yeah. I mean, there are devices out in the market now, you know, one of them for a detector that they develop for the International Atomic Energy Agency for their safeguard, which is the, you know, INPO measurement, and they do use, you know -- they're reading gross gamma, you know, and the neutron from curium, and they take those readings which they correlated to the burn-up.

So I mean, there are these devices out there, and also British device there's one out there, and these have been demonstrated.

DR. LEVENSON: let's go on.

MR. GUNTER: Okay. The second issue that we have, and I don't think this is near as significant, but it has to do with the analysis of the aircraft crash hazard. The NRC stated that detailed projections of future flight activity would be required and a license application.

DOE believed that we would be able to reasonably bound some of the assumptions, and that should be acceptable. Primarily we had concerns with, you know, how accurately we could predict future flight activities and changes in the Air Force missions and things like that. That would be a function of information we could gather from the Air Force.

And also it's fairly speculative as to what future missions or future aircraft might be. So that was our only concern on that one, but we did agree that we would provide the NRC a revised analysis on this topic.

DR. LEVENSON: Does an aircraft crash represent the accident of significant public consequences?

MR. GUNTER: I'm going to ask Dennis Richardson to help me on that one, but I don't think so.

DR. LEVENSON: I mean, your fuel is either inside a hot cell or inside a cask.

DR. GARRICK: And there's no stored energy to speak of, right?

DR. LEVENSON: And there's no stored energy to dissipate.

MR. RICHARDSON: Yeah, at present the analysis we have done on this -- of course, this is a hazard that we have investigated -- was done on the conceptual SR design, and based on the data we had at that time, we were able to screen out the aircraft hazard because of falling less than ten o the minus six.

Now, we certainly will take another look at this. We're gathering more data, a couple more years of information from the Air Force, and once we have the actual -- a little more information as to the footprint and the layout of the structures, we'll know better whether or not we can screen this out.

If we can't screen it out, and this will get to your question now, then we would have to come up with a safety design basis that would have to be also incorporated into, say, the waste handling building, for example, to insure that it could take such a hazard without any unduly consequences to the public or any major release.

So right now it's screened out. If we can't screen it out, then it will contribute a design basis to the SSCs for the preclosure period.

DR. GARRICK: Have you done enough analysis to have a sense of what the impact is? I would guess this is not an issue.

MR. RICHARDSON: It shouldn't be an issue because, as I said, if we do have to consider this hazard, we would have design basis on, say, the structures and buildings such that it could take such a hit.

Now, we haven't done that yet, but I believe that would be the first strategy that we would follow as opposed to assuming consequences and then have to calculate it.

And the bounding approach that Tim talked about, we believe that's a fairly good way to go. In the future obviously if any of our assumptions or something would show, say, something new comes up with the Air Force that would, say, violate or exceed the bounds that we assumed, then we certainly would follow, you know, the 6344, which is your 5059 Safety Federation would consider it, and we would simply just follow that as a consideration of a change to our licensing basis.

MR. GUNTER: Okay. Page 13 summarizes the agreements that we came to with the NRC. There's a total of nine agreements. I think by the previous discussion we've had here we've covered most of them or at least led to them, but we agreed that we'd provide a comprehensive aircraft hazards analysis and to meet again with the staff to discuss that analysis. We would provide a comprehensive tornado miss analysis, hazard analysis. We'd update the procedure for identifying SSCs important to safety and waste isolation, and we would provide that to the NRC.

We --

DR. LARKINS: Does that agreement include the use of expert panels in the classification of SSCs important to safety and any special treatment requirements?

MR. GUNTER: I don't think the agreement specifically addressed that, but that's probably in that procedure.

Paul?

MR. HARRINGTON: We talked a lot during the technical exchange about that specific issue and the potential for doing that and agreed that we would use them as appropriate. The concern that I had was going beyond what we had done in the commercial nuclear industry, where particularly when you got down to piece part classifications and even component classifications, that was done by an engineer in the design control process, and it went through the reviews by other disciplined engineers, independent reviews and the disciplined managers.

The way I initially read that particular question from the staff, it seemed as if they were looking for us to do independent review groups for all classifications, and they agreed, no, it was not their intent.

So, yes, we did discuss that, and we're really looking at adopting a fairly similar process to what we had done in the commercial nuke side.

DR. LARKINS: Okay. Because I know recently in the special treatment requirements for SSCs and the safety for reactors, the use of expert panels was very important because the PRAs only covered five or ten percent of the components, and the others were covered by the panels.

MR. HARRINGTON: Okay.

MR. GUNTER: Other agreements we reached was to provide them with a copy of our guidelines. We're developing guidelines for conduct of the ISA.

We'll provide them an update to our preclosure criticality analysis process report, and in that we would include an approach for verification of fuel assembly burn-up.

We would provide them an update of the waste package design methodology report, and this is to address some of their issues on the waste package drop analyses and also variations in the material C-22 and weld filler material.

And then the last on that page is the waste package operation fabrication process report. There are two agreements to provide them updates to that, and that was to address their questions on NDE methods and also mechanical properties of disposal container.

DR. LEVENSON: Back up on the second bullet, the comprehensive tornado missile hazards analysis, is that related to some sort of risk?

You know, the question comprehensive, this is so different than reactor plant. There's not much. It's hard to imagine what a tornado can do that generates a public risk.

CHAIRMAN HORNBERGER: A tornado can do plenty.

DR. LEVENSON: What?

CHAIRMAN HORNBERGER: It will wreck the nearby mobile home park.

DR. LEVENSON: I know, but what can it do to this facility which generates a public risk?

MR. HARRINGTON: During the technical exchange the focus was really on the transporter from the time it leaves the waste handling building until it got to the underground, and what we had done was looked at the missile from a ten to the minus sixth earthquake event -- yeah, tornado event, but then we said the transporter is only exposed for a small portion of the year because it exits the building and takes a few minutes to traverse over the underground.

So we had effectively screened out tornado missiles form that transporter consideration. We agreed that that wasn't really appropriate. So what this is intended to do is to say that we will go back and look at what sort of missiles would be expected to fall within a ten to the sixth overall period for the duration that that transporter is actually exposed to missile hazards and use that.

DR. LEVENSON: Oh, okay. I would not define that as a comprehensive tornado missile analysis.

MR. HARRINGTON: Okay.

DR. LEVENSON: Okay. So it's a very limited one.

MR. HARRINGTON: It's comprehensive enough to capture the --

DR. LEVENSON: Yeah, but it's only the transporter between the building and --

MR. HARRINGTON: Yeah, okay.

MR. RICHARDSON: Well, just to add to that, the waste handling building -- the rest of the time, of course, the waste package is within the waste handling building, and of course, the waste handling building will have a design basis to be able to adequately take care of the missile.

DR. WYMER: Is there not a waste storage building in addition to the waste handling building?

MR. RICHARDSON: Yeah, this is where it's received and taken out of, you know, the canisters, if so, and loaded into the waste handling package and sold before it's delivered underground.

DR. WYMER: Or if they have to be retrieved for some reason, there's a place to put them back.

MR. RICHARDSON: Yeah, you're absolutely right. But that wouldn't be the waste handling building. That would be a separate --

CHAIRMAN HORNBERGER: Just out of curiosity, this is ignorance on my part. When they move fuel around at a reactor, do they have to go through and do a tornado analysis, missile analysis?

MR. RICHARDSON: I'll take that. All credible hazards have to be also analyzed at a reactor, and it depends on the set-up system and what their operational practices are.

So in a sense the answer would be yes.

CHAIRMAN HORNBERGER: I guess my point was that people have gone through such analyses. I mean, this isn't a first of a kind.

MR. RICHARDSON: No.

DR. GARRICK: Yes, they've gone through such analysis. No, this is not a first of a kind.

MR. GUNTER: Okay. Page 14 is a couple of meeting follow-up items in addition to the formal agreements. We agreed that we would provide comments to the NRC on their raft staff position that they presented at the meeting. This was the one on the risk significance categorization of the SSCs important to safety.

And then DOE would also provide a white paper that laid out the details of the quality classification and grading methodology.

Okay. In summary, we felt like this was a beneficial meeting for Department of Energy. We gained an increased understanding of NRC's staff expectation, the level of detail to be included in the license application, an we also obtained valuable feedback on the acceptability of DOE's integrated safety analysis methodology and a proposed differentiated approach for information that we would include in the license applications for construction authorization and then for receive and possess.

DR. LEVENSON: I've got a question on the previous slide, which said DOE will provide formal comments on the NRC draft staff position on risk significant categorization of structures, systems, and components.

What does that mean? I mean, were you given a list of things that were risk significant or what does that mean?

MR. GUNTER: Well, we were provided their staff position paper on this topic, and we agreed that we'd review that, and if we had any comments on that paper, that we would formally submit them to the NRC for their consideration.

DR. LEVENSON: Is this a staff paper specific for Yucca Mountain or is it the standard one for reactor systems? Do you know?

Can somebody from the NRC staff tell us?

MR. JAQANNATA: Banad Jaqannata, NRC.

This position was a staff position specific for the Yucca Mountain project. DOE had made a proposal on this one. We had a lot of discussions and exchanges between two of our positions, and this one kind of stayed our position and concerns on DOE's proposal, which was discussed in technical exchange and came to common understanding during the meeting.

What DOE will be doing is giving their formal response, and we also have an agenda item to finalized our position by the end of the year. This one is specific for the project.

MR. GUNTER: Any other questions for me?

DR. HINZE: I just want to make certain that I'm clear that the concerns that the NRC have regarding seismicity during the preclosure period are being handled under the structural deformation KTI?

MR. GUNTER: That's correct, and there is work going on to address that and another topical report.

DR. HINZE: Right.

DR. LEVENSON: You're not going to have any earthquakes during the preclosure period. So --

PARTICIPANT: If you say so.

DR. HINZE: Well, but they didn't move it here. They moved some things here.

DR. LEVENSON: John?

DR. GARRICK: I'm fine. Thank you.

DR. LEVENSON: Ray?

DR. WYMER: Are we to the end of the summary? Okay.

MR. GUNTER: That's the end of my

presentation.

DR. WYMER: I did have a question, and you're going to quickly understand why I asked about additional storage space.

Is it part of the preclosure safety analysis considerations that there's a possibility of theft by a proliferator of some of this spent fuel? And if it doesn't belong in this preclosure safety study, where does it belong?

MR. GUNTER: There is a separate event, well, an effort going on for the safeguards and security issues out of our headquarters office. I'm not that familiar with the details of that process, but I know that it's being addressed separately from the preclosure technical exchange.

DR. LEVENSON: Normally security things are done differently and not in public meetings.

DR. WYMER: Well, this was a general question. I wasn't asking for your secrets.

DR. LEVENSON: Any other questions?

DR. WYMER: No.

DR. LEVENSON: NRC staff?

DR. LARKINS: Yeah, this is a quick follow-on to the other related to categorization of SSCs. Is there a general agreement on the methodology for importance measures or the use of importance measures?

MR. GUNTER: Do you want to take that?

MR. HARRINGTON: If you want to expand on that, but I think the short answer is yes.

MR. GUNTER: And Paul's going to discuss in a lot more detail a lot of this, you know, these specific topics.

DR. LEVENSON: Okay. You're back just in time, George, for your questions.

CHAIRMAN HORNBERGER: No, I think I've asked mine. Thank you.

DR. LEVENSON: Anyone else have any questions?

MR. AHN: I had one.

NRC staff, Tae Ahn.

Regarding your question on the relation of preclosure and the post closure system, that everyone comment in the area of waste package performance. The personnel who are working in the preclosure assessment and post closure performance assessment are the same, in the DOE and NRC side.

For instance, in the preclosure performance assessment, (unintelligible) the waste package is most important source of radionuclide release. On the other hand, in the post closure period, the corrosion of waste package the most important source of the radionuclide release.

To assess both failure mechanisms, processes, welding is the common issue. Therefore, both preclosure and post closure performance assessment, the welding design is the key element, the same personnel working on that issue.

MR. GUNTER: Okay. Any other?

(No response.)

DR. LEVENSON: Okay. Thank you.

Oh, I'm sorry.

MR. SINGH: Just a follow-up John Larkins' question about the SSE. You said, yes, we do have agreement. Is that what was mentioned in the June 13 draft correspondence to you between the NRC staff and the DOE?

MR. HARRINGTON: It was mentioned in that correspondence, but we also discussed it during the meeting.

MR. SINGH: Okay. Thank you.

MR. GUNTER: Okay. If there's no other questions, I'll turn it over to Paul Harringston who's going to go into more detail on the preclosure plans that we have. I'm going to swap pots with you.

MR. HARRINGTON: Hi. I'm going to talk to three major topics here today. The first one is the preliminary preclosure assessment. It's a product that we have already issued, and we have just issued an ICN-3 to Revision 0 to it, interim change notice in late June of '01. So this is really the product that the first third of this discussion is going to talk to.

The second third will be to talk about the integrated safety assessment that we will do for a license application itself, assuming we go through a site recommendation.

And then the third is to give you an update on what we've done in the engineering activities recently.

So with that preliminary preclosure safety assessment, this is not an ISA. It's not a quantitative evaluation. This was really a qualitative evaluation. It was done to support the site recommendation suitability evaluation. As I said, we did it in 2000 and updated it just recently.

It used a lot of the elements though from an ISA. it did hazard assessment, fault trees, event frequency assessments, and had some consequence estimates. So we recognize that it's not to the level of detail that will be necessary to support a Part 63 submittal, but we think it was an appropriate product for use in a site recommendation consideration.

Well, in doing that, we ran through several different activities in there. We first identified the applicable external and internal hazards. Then we did an analysis of the design basis event frequencies and consequences of those, then identified the SSCs that would be important to safety, and then developed strategies for criticality, fire protection, rad protection and low level waste management.

For the external events hazard analysis, we had a generic checklist of 53 event sequences or categories. We screened for the potential at the site. Tsunamis, things like that would immediately get screened out.

PARTICIPANT: It depends on how much global warming.

DR. GARRICK: You might have a magma tsunami.

MR. GUNTER: And the San Andreas Fault activity and that kind of stuff, which is Arizona beach front property.

And event frequency. In doing that, we

grouped these events that remained into a series, the loss of off-site power, seismic vibrations or displacement flooding, winds, tornados, and missiles.

For internal event hazard analyses, we had a generic list of six categories. Collision crushing, that would include dropping or mishandling components, chemical actions or flooding, explosions or implosions, fire, radiation or fissile issues, and thermal.

We got these out of the system safety analysis handbook and guidelines for hazard evaluation. So we tried to use a common industry guidance to do this work.

We did a systematic analysis of each of these operational areas to identify the presence of the hazard, and then used them as initiating event input for the event sequences.

I'll repeat the category definitions out of proposed Part 63. For Category 1 it's those that are expected to occur one or more times during the lifetime of the facility, the preclosure period.

Now, you'll see all through here our analyses are based upon a 100 year preclosure period. You'll hear us also talking about potentially extending that up to 300 years. If we do that, we recognize that that then needs to be factored into reclassifying, potentially recategorizing events into Category 1 or 2.

Likewise, sine Category 2 is -- in fact, we'll flip to the next page -- Category 2 is one in 10,000 over the preclosure period, then that also would have some effect if we were to extend that preclosure period.

But for the purposes of the analyses that we've done to date and would currently expect to do in the event of a license application, we're using 100 year preclosure periods.

At page 10, and I'm trying to get through this first part quickly because I think there will be a lot more questions a little later, and we have only an hour. We did the event trees to identify event sequences, then estimate the frequencies of those events and the probabilities using industry data, licensee event reports.

We did that using the maximum throughput expectations for the facility. Nominal is 3,000 MTHM per year. The CRIMS requirements document, the highest level document requires that we have the ability to throughput I think it's 20 percent greater than that.

DR. GARRICK: Now, when you talk about maximum throughput, are you also talking about maximum inventory on site or in the facility?

MR. HARRINGTON: Dennis, you can answer that.

MR. RICHARDSON: Yeah, I'm not quite sure I get the gist of --

MR. HARRINGTON: Well, are the pools loaded up entirely?

DR. GARRICK: Yeah, he's talking about the maximum throughput rates for operations. Is the facility at its capacity for receiving spent fuel?

MR. RICHARDSON: John, I'm not quite sure of our assumptions in that areas. Obvious this is going to be dependent somewhat on the operational considerations, whether we're wet, dry, in between, that kind of stuff. But as far as the actual initiating event frequency, say, for number of lifts, I think that's just dependent on bounding the assumed throughput for a year.

DR. GARRICK: Because some events it would make quite a difference between whether it's just a throughput condition or you have a full house.

MR. RICHARDSON: You mean like if you drop something where it's going to -

DR. GARRICK: Yeah.

MR. RICHARDSON: Oh, okay. Well, a drop --

DR. GARRICK: An aircraft impact.

MR. RICHARDSON: Sure. Like on a drop assumptions, a drop in certain areas, we always would assume that it -- like if there is the possibility of another basket there or something like that, that that would occur. So we try to what I would say be reasonably conservative in terms of those assumptions given that you have a drop in terms of what events may follow from that.

MR. HARRINGTON: Based upon then the results of the frequency and probability, that determines whether or not things fall into the Category 1 or 2 or beyond design basis event.

Now, page 11 is the summary of the event sequences for the Category 1 internal DBEs. You'll notice they are all driven by assembly transfer issues. That's simply because of the large number of individual fuel assemblies that the surface facility would have to handle. That's the reason for those falling into the expected events, the Category 1s.

DR. WYMER: Are these results based on spent fuel handling experience?

MR. HARRINGTON: Yes, yes.

DR. WYMER: The frequencies per year?

MR. HARRINGTON: Yes, yeah.

Let's see. Page 12 then gives the dose summaries for the sum of those Category 1 events, and as you can see in the third column, the doses are very low compared to allowables.

We put a series of dose standards in there. The upper ten MR that I have cited to Part 20 is an ALARA issue. One hundred MR was the limit from that.

Now, we also did the worker doses at 100 meters, and we did the public doses at seven miles for a boundary from the waste handling building itself, and five miles from the ventilation shaft penetrations through the top of the mountain.

So I wanted to get across that that's not at an 18 or 20 kilometer boundary.

DR. GARRICK: So the public that went to get their chest X-ray that day got the worst of it.

MR. HARRINGTON: Yes, yes.

(Laughter.)

MR. HARRINGTON: Page 13, page 13 is a summary of the Category 2 preclosure DBs for off-site doses. These were treated individually, and you can read down. The doses themselves are quite small.

These doses assume that the HEPA system is working. If the HEPA system is not working, then the maximum dose is on the order of two rem, still appreciably under the five allowable.

Page 14 then discussed the preclosure strategies for design of the facility.

DR. LEVENSON: Excuse me a minute. Back on the other --

MR. HARRINGTON: Yes.

DR. LEVENSON: What assumptions generically were made in these drop accidents? How did you get from a drop accident to a release fracture?

MR. HARRINGTON: Dennis, you can answer that.

MR. RICHARDSON: Well, some of the drop accidents, first of all, some of the drop accidents we would assume obviously that the container would not be breached, and so there would be no release fractions, no radiation doses.

For those where we would assume breach where we're dropping their fuel, release fractions are based on precedent calculations that are done for, you know, their commercial fuel and stuff.

DR. LEVENSON: I know that.

MR. RICHARDSON: Now, for the Navy fuel, of course, the Navy does their own calculation, which we will utilize because there we assume a breach.

DR. LEVENSON: My point is for the probability of drop, you're using data, not calculations.

MR. RICHARDSON: That's right.

DR. LEVENSON: There's an equal amount of data on actual release fractions because every drop that has occurred has been measured, and I think without exception they've been orders of magnitude less than the calculated releases.

MR. RICHARDSON: Yeah.

DR. LEVENSON: The calculated ones was done before we had all of this experience. My question is: why are you using real data for the probability and assumptions for the release fraction?

MR. RICHARDSON: Well, on this preliminary analysis, we admittedly are, for the dose calculation, are using what I would call conservatively bounding type of calculations for that end.

And you're right. For the probabilities and stuff we will use actual data and try to use mean values for establishing the categorization.

MR. HARRINGTON: Good comment. Thank you.

MR. RICHARDSON: And where we will end up when we actually do the LA I'm not quite sure yet. Obviously if we have what we consider a lot of margin and fairly defendable, conservative ways of calculating that dose, we might go that route from the defendability viewpoint with the NRC, but I'm not sure yet.

DR. LEVENSON: I'll just give you my personal opinion. I'm of the old school, and that is actual measurements and what has actually occurred is a lot more defendable to me than somebody's calculation no matter whether it's more or less.

MR. HARRINGTON: I understand.

DR. LEVENSON: The calculation is different than the real world. So the real world must be wrong, right?

(Laughter.)

MR. HARRINGTON: Okay. On page 14, the basic strategies for each preclosure operation. The containment confinement augmented by prevention, what we're really talking about there is as Dennis said a moment ago. We really expect a number of the canister designs not to breach on a drop. We think the drop and breach is incredible, but you still would want to minimize the potential for having a drop. There's no need to exercise that canister integrity by doing the drop.

Conversely, the second sub-bullet under that, prevention augmented by containment/confinement relative to bare fuel. You would expect to have a system that would not allow a drop, minimize the ability to do a two block or to somehow have a drop accident, but recognize that you have defense in depth in that you have a containment or confinement system should you actually have a drop. That's the intent there.

Page 15, in the PPSA, the last several chapters of that also provide our strategies for criticality, rent protection, fire protection, and rad waste management. We used the NRC Reg. Guide 371 in part for development of the criticality strategy.

Fire safety, we'll look at both rad safety of the workers as a consequence of that and also the nonradiological health and safety of workers.

Rad protection, certainly ALARA, the management support of that program, the components and system design that will enhance the ALARA program.

Page 16, we also in this interim change notice took a look at the potential effects of the several various modes of achieving lower temperature post closure design solutions. I'm not sure of the extent to which you've been briefed on that. A little later I have in the third third of this a summary of five different scenarios that are captured in the science and engineering report describing different ways to get to a low temperature waste package and rock and also what we did in the SSPA that Rob Howard briefed you on yesterday. That was yet another.

Suffice it to say there are a number of different variables that we have available to us, including waste package spacing, degree, and extent duration of preclosure ventilation, et cetera, that can allow us to do thermal management, and in looking at those in this ICN-1, we didn't see that any new events were introduced.

It also took a look at the potential for surface storage, acknowledged that that would potentially introduce some events that had not been assessed in the earlier versions that did not have a storage pad that would be used for the aging if we were to do that.

But we also believed based on some very preliminary work that nothing would be introduced in that mechanism that would exceed the doses that have been evaluated. So that will be done further in an ISA.

Go ahead.

DR. LARKINS: A quick question on your fire analysis.

MR. HARRINGTON: Yes.

DR. LARKINS: You agreed upon a methodology. Are you going to use a screening or have you an approach for doing your fire? Can you say something about how you're going to do your fire analysis?

MR. HARRINGTON: I don't know if we have defined that approach yet. Dennis, do you know?

MR. RICHARDSON: No, we obviously will look for all fire hazard areas, potentials for that. We'll do some screen from the viewpoint of how much inventory is required to really worry about it. If we have different issues in that line, we'll follow commercial precedence in terms of probably setting like technical specifications for inventories and things like that that would be a problem.

DR. LARKINS: Maximum heat loads.

MR. RICHARDSON: Right. So we'll try to follow precedent as much as possible in that area

MR. HARRINGTON: And we do recognize the

NUREG 1567, the SRP for dry fuel storage facilities as guidance that's available to us in this when it gets appropriate to use.

DR. LEVENSON: On your last bullet, what are the new design basis events that come into being if you have a low temperature mode?

MR. HARRINGTON: That was really associated with the storage pad that wasn't in the earlier -- it's not the base case design now, but if we were to decide to do thermal management through aging of fuel and put a storage pad out there that had a capacity of up to 40,000 MTHM, you'd been looking at tornadoes, potentially aircraft, missiles, possibly others associated with that that you just don't have on the base facility.

DR. LEVENSON: It doesn't come directly from the lower temperature.

MR. HARRINGTON: No.

DR. LEVENSON: The lower temperature changes the facility.

MR. HARRINGTON: Right.

DR. LEVENSON: Okay.

MR. HARRINGTON: Right. So in summary, on page 17 --

DR. GARRICK: You're not going to do

blending?

MR. HARRINGTON: Oh, no, I didn't say that. There's --

(Laughter.)

MR. HARRINGTON: There's still a 5,000 MTHM capacity pool inside the building that regardless of whether or not we would do the aging on the outside facility, there's still the capacity to do the blending inside.

And as we focus more and more on thermal management issues, it appears less and less like we would not need to do some sort of thermal management by blending.

DR. GARRICK: I see.

MR. HARRINGTON: The PPSA was done to support our recommendation. A similar process to an ISA, but less detailed, more quantitative. Both off-site and worker doses are well within Part 63 bounds and came up with preclosure strategies. Don't significantly or don't see those expect to significantly change by some of the low temperature alternatives.

Now, moving to the integrated safety analysis, this does need to be a much more quantitative discussion than we had done in the PPSA.

I'll talk through how that's risk informed, not absolutely risk based. There are some deterministic features in this.

It takes advantage of industry precedent, and discuss some of the products that will be associated in development of that.

On page 19 --

DR. GARRICK: Is this the same thing that when the NRC talks about integrated safety analysis?

MR. RICHARDSON: Yes, we sure hope so.

MR. HARRINGTON: Same terminal.

MR. RICHARDSON: Until it's changed.

(Laughter.)

MR. HARRINGTON: We really are very supportive of the technical exchanges and other interactions just to make sure that answer can be yes. We think it is yes. We really think we now have agreement with NRC staff on the approach that I'm going to walk through here in terms of identification of classification and doing the dose analysis.

So this line is really our interpretation of the off-site dose criteria that Part 63 gives us, and I'll use the pointer here.

In fact, have we given this to you before? I really don't remember doing it.

Okay. This is the summary discussion. This line represents the Category 1 events. The frequency is more frequent than one in 100 years.

This is the ten to the minus six, the ten to the minus four representing one in 10,000 from the 100 year period out.

This is the five rem per event, total effective dose equivalent. That's the limit for category 2. This is the 25 mr per year total effective dose equivalent for Category 1, and this line is 100 mr per year that comes from 10 CFR 20, though these vertical lines represent the one year point where we took it directly down. We wouldn't extend it down to get a greater dose for a less frequent period.

So this represents a per year dose, then the one year period.

So this is really the graphic representation of dose. This red line really represents the bounding allowables. To the left is okay. To the upper right is not.

Page 20, just as the definition out of Part 63 for an ISA is. In the interest of time, I'll skip over it.

Page 21 is a summary graphically of the

process that we'll use to do an ISA. The next several pages, eight or nine pages, will go through most of these blocks and give a text discussion of them, but just to grasp the overall flow, we start up here with the design, the site characteristics, and the operational features.

We'll go to both internal and external hazard identification, and then move down to identification of the sequences. Define the event trees and fault trees, and then do a screening of the frequency assessment.

If it screens out to be more than one in ten to the minus six or less frequent than one in ten to the minus six, then it moves out to the side and is considered a beyond design basis event, and that then is picked up later. We'll come down to that in a moment.

If it is not a beyond design basis event, then it goes into the categorization. Is it Category 1 or 2? And then the consequence analysis, and then based on the results of those two, does it meet the limits that I gave on the previous slide?

If it does, then that's an acceptable design that will feed into the design, evaluation, and support, be captured in design criteria in the system description documents. It will go onto the Q list.

If it does not meet those regulatory criteria, then we have to look at what can we do to prevent or mitigate it and feed that back into the design process in the upper left.

Now, a couple of other blocks. We do incorporate regulatory precedent. That was in some of the earlier slides, and the PPSA certainly will do that for the ISA.

Also, on the beyond design basis event, we talked at length with the staff about the flatness of the probability distribution functions associated with that. If something screens out just beyond the ten to the minus six, but yet you have a very low confidence that that is really where that event is, we will most likely continue to consider it.

But if you have a reasonable expectation, a narrow enough EDF that something is beyond ten to the minus six, then we'll consider that as screened out.

These bottom set of products would then be used to feed a potential license application.

Page 22 looks at the set of products that would go into an ISA. Boy, that doesn't show up too good for some reason.

The left-hand column are those associated with the hazard identification. You'll see in parentheses under some of them document numbers, ANL-MGR-SE. That's an analysis that was done on external hazards.

We also have done one on internal hazards. The next body represents work supporting the DBE categorization. The next supports consequence analysis; the next, identification then of those SSCs that are important to safety and prevention or mitigation features.

And then finally the ISA analysis and set of products supporting it.

We're using guidance from regulatory precedence and current development. There's a series of several that have gone on and are going on. We're following the Exelon pre-application on the pebble bed reactor just to make sure that we're getting the NRC's latest thoughts and can incorporate them.

DR. WYMER: Graphite fuel in the repository?

MR. HARRINGTON: Well, actually yes. The (unintelligible) fuel is graphite.

DR. WYMER: Is that going in there?

MR. HARRINGTON: Yeah. It'll have to go

to a repository.

DR. WYMER: Yeah. Well, yeah

MR. HARRINGTON: So let's see. Page 24, from experience we'll pick up identification of potential hazards and then the estimates of those event frequencies and probabilities, and yet, this would be a wonderful place to pick up actual dose results rather than simply calculations for release.

One thing we wanted to point out. We've gotten agreement, we believe, with the staff regarding the use of mean values for the frequencies for the use of binning. Along with that then is the discussion of the probability distribution function.

Industry precedence for external events, tornado and wind -- I won't read through them all -- aircraft hazards and floods, there are a number out there that we're making use of.

It brings us to page 26, and this is our proposed criteria then for a risk informed classification of SSCs. Now, the additions here are this line and these pair of lines. What we've done is said if the regulatory limit is 25 mr per year, then two orders of magnitude less than that we think represents a significant step.

Also, extension of these lines at 25 mr

and 100 mr into Category 2 space represent reasonable numbers to use for classifications.

The reason we did this is previous versions of this slide that we've shown over the past couple of years didn't have these sets f lines in them, and we had sort of a general gradation from quality Level 1 through quality Level 2, 3, and then the commercial quality, and it was very difficult to understand what our real rationale was in trying to do these classifications.

So we identified that as the discriminators -- next slide, please -- and then assigned these regions to them, and this region would be unacceptable. This would be -- well, and associated with quality Level 1; this region associated with quality Level 2 and 3 and 4.

Now, the reason it looks like this, we're recognizing that there's a difference between the Category 1 and 2 events. So there's a progression as you move across Category 1 and a progression as you move across Category 2 events based on those consequences from things that would be considered commercial quality to quality Level 3, then 2, then 1.

So page 28, QL-1, effectively equivalent to safety related for reactor space. For QL-2 and 3, we would grade that QA program, and for commercial quality, we would just apply a normal commercial quality program to it.

Now, the question has been and honestly continues to be what is grading. The staff wants to know that. Well, so do we.

We haven't yet put a lot of time into trying to answer that question yet though because our focus really as an organization is on site recommendation. That's where the large majority of the program resources are going. So to at this point provide the resources to define exactly what do we mean when we say the QL-2 and 3, what are we going to grade out versus include, we simply haven't put the time into that yet.

Let's see. The important to safety, this is just a note, and Part 63 is really more inclusive than safety related was in Part 50. Things that in Part 50 space had not been safety related, like fire protection or seismic two over one, safeguards and security, some of those things we had sort of built up a methodology for dealing with them outside of the safety related space.

This definition is a little different. It has caused us to rethink how to approach some things versus what we did in commercial power space, but we think we're getting there. We think that we and the staff have a way that we can work these definitions. So we're okay with that.

Let's see. Page 29 really was a fairly confusing discussion at the technical exchange for those that were there and remember that. What I'll say simply is that we were trying to be too detailed in exactly how we went through the process of determining the consequence of the one offs, where we would remove a system structure component and then look at the effect on dose as a consequence of removing that.

We walked through a process that looked at the frequency with the item in versus out, and a lot of other stuff. The only thing that really mattered though and the thing that will drive the actual classification of the component is the resultant change in dose.

So this simply says if you have an event tree that includes a series of SSCs and with all of those functioning, probability and dose fall somewhere on this chart. Let's say here, and then you take away one of those SSCs. Where does the dose end up?

And if the dose ends up in this space,

then that SSC that was removed has to be a quality Level 1 SSC. If the dose ended up in this space, it would have been a QL-2. A QL-3; then if you stayed in that space, then that SSC can be a commercial quality component.

So that's the final result of hours of attempt at discussion. So I think the staff agrees with this.

In summary then, for the ISA process on page 30, we think it's --

PARTICIPANT: Can I ask a quick question?

MR. HARRINGTON: Yes.

PARTICIPANT: Just understanding, so you've agreed on the categorization, but you haven't yet agreed on what special treatment requirements you would have for those categorizations?

MR. HARRINGTON: Yes, specifically the grading that goes on in QL-2 and 3. That's right.

Okay. In summary, on the ISA process, we think it's risk informed. It does a hazards assessment. It quantifies the frequency of event sequences, then categorizes them into Cat. 1, 2, or BDBE.

We do have to do the uncertainty assessments to -- particularly for those that fall into the BDBE category, and then do the classification of the SSCs based upon the contribution of that SSC to dose.

We will use industry precedence, experience, and data.

With that I'll move into the recent engineering developments. This is really a fairly short discussion. As I said earlier, we are spending most of our resources addressing thermal issues and NRC key technical issue agreements and NWTRB focus areas, and those are driving very heavily by post closure issues.

So we've done very little in the way of changing preclosure design solutions. I know last time I talked to you there was concern about blending and the additional handling that would go along with that. I believe I said that we were reconsidering some of those decisions like a wet versus a dry.

We have not yet concluded any of those studies. That may happen toward the end of this fiscal year, which is next month. It may extend somewhat into the next fiscal year.

But given that our focus is really on thermal management, and even in design space, that's where we're putting our efforts.

In response to a number of the comments we got from the public at some of the hearings and also questions we had gotten from NRC staff about what is this design, nobody can pin it down. It seems so flexible as to be nothing short of vaporware. We decided we need to do a better job of identifying really what we think the fixed attributes are.

We think this thing does have some fairly specific features to them, to it, and it has some operational variables that can be moved, but to convey it as something that's wholly unsettled, we didn't think did justice to it.

So we did issue this spring in May the science and engineering report. It identified this higher operating temperature mode, and it had five example lower operating temperature modes. Those are simply examples with the variables that we have available to us.

You could modify them and do things differently as the PA folks did and the SSPA, low temperature operating mode. I don't know if you saw, but that was a little different way of getting to that.

Of the five that were in the S&ER, four of them kept the waste package maximum surface temperature below 85 C. after closure. The fifth one kept it below 96 C. after closure.

TSPA for SR was based upon the higher temperature operating mode. As Rob said yesterday, they've done the SSPA, and that used the lower temperature operating mode that they go to through a little different mechanism.

DR. LEVENSON: Do all of the low temperature modes require either blending or long aging?

MR. HARRINGTON: They all require -- the short answer is no. There's a table and a couple of pages, and I'll talk to that when we get to.

So present design methodology, we think we really have a single design, but it does have a range of operational variables. We can get different thermal responses from that, and in the TSPA and ISA that would support a license application, those would be bounded by a specific thermal operating curve.

Whether or not it's at the lower end of the potential operating ranges or at the higher end, I'm not sure that we have made that decision yet, but whatever it is that does go into TSPA LA and the ISA for an LA, it would be based upon whatever the final selection of that thermal mode is.

Now, this is not the chart I was referring to a moment ago. For the single site recommendation design, this is a series -- let me get to the right page here -- of what those more or less fixed features are.

Seventy thousand MTHM, the law provides that limitation on what the NRC can license at this point.

Emplacement rate, 3,000 MTHM a year.

Emplacement period is about 23 years.

Now, on a follow-on, if we do go to the extended aging, then the overall preclosure emplacement period would be extended by the length of the aging associated with it. There is the caveat on that.

Waste package emplacement is we do have some packages that are hotter, no getting around it, but conversely, there are some that are much cooler. The defense high level waste packages, the naval spent fuel packages, the majority of the DOE S&F packages are all relatively cool.

So the expectation is in trying to manage a thermal profile to intersperse higher and lower temperature packages, we're still looking at large, horizontally emplaced waste packages on pallets.

The outer LA-22 and inner 316 nuclear grade shells, that's about 11,000 of them at the capacities that we were looking at. One of the earlier questions in Tim's discussion was were we looking at having to handle assemblies differently if we didn't end up being able to credit burn-up.

Well, no, we wouldn't handle the individual assemblies different, but the result would be that if we're not able to get that burn-up credit and, therefore, were not able to load that number of assemblies in the waste packages, consequently we'd have more waste packages.

Conversely, if we -- some of the approaches go to smaller capacity waste packages for thermal management reasons independent of criticality issues.

Let's see. Eleven, point, eight kilowatt maximum is still the limit on those waste packages; still five and a half meter drifts based on 81 meters with continuous titanium drip shields.

Now, you'll also hear about the potential for going to stand alone drip shields. The economic payback is when the waste package spacing gets to about three meters. It may be more economical simply to have individual stand alone waste packages where each one would have a short skirt extending down the ends rather than trying to bridge all of that empty space with drip shields.

Potential range of operating variables. None of these are intended to be used as a group. For example, you wouldn't have all of the lower temp. 1s together or necessarily all of the upper, but for each variable, this is the range for that variable that we're looking at, given the different operational variables that we're looking at.

So if you have questions on that, we can go through them individually. I'd rather skip to the next one.

And this will get to Milt's question.

DR. LEVENSON: One quickie thing. On the assembly enrichment number, your five percent, this is relevant only to commercial spent fuel, right?

MR. HARRINGTON: Yes, yes. We recognize the Navy and DOE have fuel that are far higher than that, yeah.

Okay. Page 37, and actually 38 continues it, lists the various options. The first column, actually second column is the S&ER design, base case design in there. It was the higher thermal operating mode, and then the five scenarios that were also captured in the S&ER.

And the two right-hand columns are the high temp. and low temp. operating modes in the SSPA. The question had to do with do all of the scenarios involve extended aging, and there's another part to it that I don't remember and --

CHAIRMAN HORNBERGER: Blending.

MR. HARRINGTON: Right, right. For the blending, we could conceivably follow number two, scenario two, D rated or smaller waste packages. That steps away from the higher thermal content that really all of the other waste packages are limited by. It would have smaller packages spaced closer together. You see the average spacing there for that scenario two of a tenth of a meter.

If we did that, conceivably we might have not to blend in part simply because you would have fewer assemblies in a package and the mix from the higher to the lower might not be as critical. But we haven't really pursued those sorts of questions for these scenarios other than to have done this amount of work to see was it even possible to get to the 85 C. that had been identified as a target value for waste package temperatures.

CHAIRMAN HORNBERGER: Refresh my memory. Does the natural ventilation require design? I mean the mountain breathes naturally. So I understand the zero in some of these, I guess.

MR. HARRINGTON: Oh, that's number of years of natural ventilation. What that really means is closure at that point.

CHAIRMAN HORNBERGER: Ah.

MR. HARRINGTON: Yeah.

CHAIRMAN HORNBERGER: So it would remain open for 250 years, but you'd just cease the forced ventilation. Okay.

MR. HARRINGTON: Yeah. Scenario one, for example, force ventilation, the second from the bottom row, forced ventilation after start of emplacement, 75 years. Okay. You have about a 23-25 year emplacement period. So you then have 50 years of forced ventilation. Then you shut off the fans and you just let the mountain continue to act like a chimney for another 250.

A few of them, scenario three and four, and also the two SSPAs said after you got done with the forced ventilation period, you simply get closure.

Now, I have a couple of fixes to make on this table. Unfortunately something didn't work quite right when we reformatted it. So under scenario four, given that that is the extended surface aging one, the surface aging column says zero. That's supposed to be the 30, and then that column under scenario five is zero.

Then also under scenario four, the emplacement period, 243 years. That's 50, approximately 50 years. It's the 30 years of aging plus the 23 years of emplacement.

And to be precise, on page 38, scenario five, the average waste package temperature at the end is less than 96 degrees rather than less than 85. So I'm sorry. Somehow we lost a little of the formatting in these.

So if you have other questions about the low temperature options, I don't know if that's been presented to you folks before, if you want to spend some time on it now or not.

Okay. We can go on to page 39 then, thermal response. I think you saw this in Rob Howard's presentation yesterday. The reason I put it in here was really to say that if you time shift by 2,600 years approximately, the low temp. maps on the high temp. So the real issue is the thermal response in the first couple of thousand years, and that's driven by the choice of variables during the preclosure period.

But once you get out to a couple of thousand years, the further thermal response is effectively identical.

So page 40, flexible operations attributes. Operate in various thermal modes; accommodate variations and moisture characteristics.

We have seen changes in the past in the waste stream characteristics, you know. The EIA, Energy Information Agency, EIA projections on waste streams have changed over the past couple of years. It's reasonable to expect that it will change in the future.

We may need to be able to change receipt and emplacement rates and also problem detection and corrective action and be able to accommodate what it is we learned during the preclosure period through the performance confirmation program.

Page 41, to reiterate the licensing approach with respect to flexible operating mode, we would present what we believe to be a single design that has a flexible operating mode associated with it by variation of a series of parameters, but that the license specifications would bound the thermal response, and that the license application would be something that would be fixed. It would not be uncertain, if you will.

In summary then, page 42, we've identified both design and operational parameters for inclusion in the engineering and performance analyses. We think that work is giving us a basis for ultimate selection one day of what those operational variables would be chosen to be, and that the license application would be bounded by the thermal response of whatever it is we ultimately choose the license application, be it high or low or something in between.

So additional questions?

DR. LEVENSON: Yeah, I have one. Generally discussion on the hot versus cold repository seems to be focused on the post closure period. My question is: have you addressed at all -- it seems to me that to achieve a low temperature repository you either have to do significant amounts of fuel handling or the way you achieve it introduces increased risks in the preclosure period.

Has anybody assessed -- and you're going to have a lot more fuel handling, therefore a lot more drop accidents, et cetera. Has there been any assessment made as to the impact on the preclosure period of the hot versus cold?

MR. HARRINGTON: Yes, we have done some assessments of that. The ISA would have to quantify that, even the PPSA. If you look at those events that are category one, they're all fuel handling events.

DR. LEVENSON: I know. That's why I asked the question.

MR. HARRINGTON: Yeah. We recognize that that certainly increases the number of fuel handling evolutions, and even for a fixed possibility of drop per handling, then the number of potential drops would increase.

Dennis, you can elaborate on that, but --

MR. RICHARDSON: Yeah, just a little bit. The document that Paul spoke to, the Appendix A of the ICN was our best qualitative evaluation of all the aspects of the preclosure safety analysis that could be impacted by going to a lower temperature, including obviously any increased handling. It could increase the frequency of certain events, things like that.

And so we did do the best we could in terms of a qualitative --

DR. LEVENSON: I'm not asking for numbers. I'm just asking whether, in fact, that was included in your agenda.

MR. RICHARDSON: Oh, yes.

DR. LEVENSON: Because the discussion we've heard has always been on the post closure period. It didn't seem to recognize that you were increasing the hazard in the preclosure period.

MR. RICHARDSON: Yeah, and we certainly obviously would quantify all aspects of that as we would move forward to doing an ISA.

DR. LEVENSON: Okay. John, questions?

DR. GARRICK: Just a couple of thoughts. On the integrated -- this is a methodology question -- on the integrated safety analysis approach that you use and that the NRC uses, when you develop essentially the space of all scenarios, what's the significance of the design basis event?

MR. RICHARDSON: Do you want me to try that, Paul?

DR. GARRICK: I think Dennis knows what I'm talking about.

MR. RICHARDSON: Okay. Well, as you can see, if you want to put up that one slide 21 just to focus people, but anyway, John, we go through. After we do all of the hazard identification, the screening, the hazards, both internal and external, that, of course, gives us our initiating events. Each one of those initiating events we develop an event tree to identify all credible scenarios that could happen from that initiating event. Each one of those scenarios then we have to insure that if there's a dose consequence on that scenario, that obviously it's compliance with the proposed regulations, or we may try to get it into a BDBE by certain safety functions.

After that is all done, then we identify each SSC that we credited in that scenario either for prevention or mitigation and what safety function was in particular credited. That safety function then becomes a safety basis, part of the licensing basis, that would be included in Section 1 of the SDD documents that Paul talked about.

So when we get done with all this over here, the design criteria and all of that, this identification of all the SSCs and their safety bases then feeds -- that becomes -- that really formulates the safety basis that would be put into the LA. It becomes part of the formal licensing basis than for the design, and then whatever design they come up with obviously, it has to meet that safety basis, which is really the key assumptions and everything that was credited in the accident analysis

DR. GARRICK: Okay. So that's a little

different than the design basis, the design basis in the old reactor regulations.

MR. RICHARDSON: A little bit different. Obviously there's a lot of similarities yet, but of course, in the reactor basis we had a set list of DBAs that we looked at, and here, of course, we will, incorporating some of the tools of the PRA, we will look at all scenarios, insure that all scenarios -- their final dose is within the allowable, and then from that obviously define anything that was credited in that to make that dose for the free closure end.

And of course we do get some safety basis also out of the post closure end, you know, whatever is credited in the TSPA.

DR. GARRICK: Yeah, but of course, you know the reason for the question is that there's a lot of people that believe the design basis mentality is what got us in trouble in terms of not adequately considering scenarios that would lead to releases.

MR. RICHARDSON: You know, you're absolutely right. In fact, under this risk informed regulation, we don't have so many what I would call, you know, here, if you do this you're okay, like single failure criteria and things like that.

We have to really put into these scenarios whatever we feel is necessary and what we feel comfortable with in order to either mitigate or prevent going close to any dose criterion, and whatever we put in there then, the basis for that, the safety basis for that, becomes part of what you would see in safety bases in any SAR.

DR. GARRICK: Yeah. Thank you.

MR. RICHARDSON: And then that's controlled obviously by the 6344 program.

DR. GARRICK: Right, right. One other thought, and I don't know that this is to the point too much, but I'm very curious about it. One of the things, wearing another hat, that we looked at on WIPP was given the fact that these facilities have a long preclosure period, in the case of WIPP it's much shorter than what you're talking about here, but it's still at least 35 years or maybe more, maybe up to 100.

And so as a part of the preoperational activity, one of the questions that was considered by the academy very seriously was what activities could take place during the preoperational phase that would increase the confidence in the long-term performance of the repository, given that these are very unique facilities, and there's a lot of competent people around for a long time and with the emphasis that we now have on uncertainty in the performance assessment.

Has there been any consideration on the part of your team at all about activities that might be introduced into the preclosure operations that would be for the benefit of something other than just getting fuel under ground, and could be for the benefit, for example, of a laboratory, so to speak, that generates information that would increase confidence in the long-term performance.

MR. HARRINGTON: What you're describing really sounds like the performance confirmation program.

DR. GARRICK: Well, yeah, yeah. It is.

MR. HARRINGTON: And the answer is yes. We have issued a couple of versions of the performance confirmation plan just to take a cut at identifying what do we think makes sense to try and monitor; what tests should we run to give us confidence during the preclosure period that will actually get the post closure performance that we would expect to get.

DR. GARRICK: Yeah, I was generally familiar with the performance confirmation program, and I was just curious if that had advanced into a different stage or the program was still something that was moving forward and whether it's going to affect your operations, the preclosure operations at all.

MR. HARRINGTON: Okay.

DR. GARRICK: I'm thinking of whether there's activities that you're introducing here that we haven't heard about that could put more workers on the site, could affect the preclosure safety in any way.

MR. HARRINGTON: Yeah. There's a test and evaluation plan also, and the performance confirmation program is a subset of the overall test and evaluation plan. The T&EP is supposed to look at overall facility testing, start-up testing, operational testing, tech spec surveillances, those sorts of things, and the PC program is really considered to be a part of that.

DR. GARRICK: I see.

MR. HARRINGTON: So we haven't done a great deal in developing each of them. There are versions of both of those plans out there now. It's not to the level that we would know that we would have to have to support license application, but for site recommendation purposes, we think that, you know, what we've done is enough.

DR. GARRICK: And you don't see this as affecting any of your event sequences, for example, in your accident analysis for this?

MR. RICHARDSON: I hope not, but we certainly will have to investigate that linkage when we do the full quantitative ISA, and that hasn't been done yet, but it's something that we certainly will have to look out for.

DR. GARRICK: Okay.

DR. LEVENSON: George?

CHAIRMAN HORNBERGER: Paul, on your very last slide, you comment that you're going to continue to converge on an optimum design and operational parameters. Who defines optimum? How do you do that and how are you going to sort out what is optimum?

MR. HARRINGTON: I think we will define what we believe to be optimum for cost, operational duration, material handling, occupational exposure, public health and safety, and a whole suite of other considerations.

Whether or not that will generally be accepted by all parties to be optimum, it would be a safe bet to say that it won't be. So we'll have to define or defend that particular selection.

That's almost the same question that the

TRB asked me when we talk about selection between different modes to achieve a low temperature operating mode and they want to know what are the criteria that we're going to apply. We haven't really closed on that.

In one earlier presentation I had given a series of criteria. They wanted to know this past January what the relative weighting was between them. We haven't established that yet, and that's something that during the licensing process some of it will even run into operations.

But I think the thrust of your question was who defines optimum. We would expect to do that and then have to defend it.

CHAIRMAN HORNBERGER: When I went back, your slide 11 is your category one events. Okay? And so events one, four and 12 all involve assembly drops, either in a cask or in a basket or in a dryer, and they're all .2, which means that roughly speaking you would expect to see three of these events every five years of operation.

And you say this is based on actuarial data, right?

MR. RICHARDSON: Yeah.

CHAIRMAN HORNBERGER: It just sounds high to me.

MR. RICHARDSON: Yeah. Let me describe how part of that has come up with. To get on the actually initiating event in terms of giving a lift, what's probably a drop, this used a couple of years of data out of the Newport News shipping yard, and we only had two full years of data. We know how many lifts, how many drops, heavy crane operation, and we use that just to get up to get a point estimate.

You know, given the lift, what's the probability drop? That's then multiplied by the assumed bonding, the assumed throughput based on the assumed operational practice for this. Multiply those two together. That gave us our probability of drop per year for this.

Now, all of that may and will change as we get obviously more data from the heavy crane type of stuff as we really hone in on what's our operational philosophy going to be and the type of design, number of lifts, things like that.

DR. GARRICK: It's not based on commercial power

MR. RICHARDSON: No. We used the shipping yard heavy crane data on this particular one, a couple of years of data there. Obviously we have, you know, more that we will do there, but based on where we were in terms of the emphasis in budget for this area, that's what we did for the conceptual design for the SR.

CHAIRMAN HORNBERGER: Now, see, I thought as John indicated that you were basing this on actually data from commercial spent fuel handling, and I was just wondering why I never heard about any of these accidents, and it must be that the consequence is nil.

DR. LEVENSON: Well, also maybe there's different quality of cranes and operators in nuclear facilities and in shipyards.

DR. GARRICK: Yeah, the numbers look very high especially for commercial.

DR. LEVENSON: George, other, more questions?

CHAIRMAN HORNBERGER: No, that's fine.

DR. LEVENSON: Ray?

DR. WYMER: Given the requirement for retrievability, I wondered what impact that had, if any, on your preclosure plans, your design, your safety analysis. How much did that add to what you would have to have or you didn't have to have retrievability?

MR. HARRINGTON: I don't think it adds anything to the operation, the expectations for operations because for retrieval, we would expect under normal circumstances to simply reverse the emplacement process. It's capable of doing that.

Now, as far as those consequence events, I don't think you incorporated retrieval into this, right?

MR. RICHARDSON: Yeah, we really haven't yet. One thing we'll have to do obviously is show that retrievability is -- we don't do anything that would prevent retrievability through the design of this operation.

I really believe that if we get to a retrievability situation into the future, I would guess that DOE will have to prepare an LER at that point and revisit impact on any of the safety analysis and then get approval and review by the Commission.

DR. LEVENSON: I have one other question. What's the range of enrichments over this whole history and all of these various reactors, the range of original enrichment in the PWR fuel and BWR fuel? Do you know?

The context of my question is you're very concerned about burn-up, but burn-up per se is not a measure of potential criticality. It's the original enrichment minus the burn-up, and if you've had a range of enrichments just knowing the burn-up doesn't tell you much.

MR. HARRINGTON: But I also know the enrichment of the individual fuel stick, too. So --

DR. LEVENSON: Yeah, but the point is you're accepting a paper record for the enrichment, but you're not accepting the identical record for the burn-up.

PARTICIPANT: They are.

(Laughter.)

MR. HARRINGTON: I would be.

DR. LEVENSON: Okay. I'm sorry.

MR. HARRINGTON: We would think that the paper record for the burn-up credit would be as valid as for the initial enrichment and would be a suitable basis for operating the facility.

DR. HINZE: Let me try a question. We've got a TSPA from the SSPA on the high and the low temperature and operational mode. Can you tell us what a rough cut is on the preclosure safety analysis for those two?

MR. HARRINGTON: Oh, in this the Appendix A took a look at the introduction of these low temperature operating modes, and it said that other than the potential introduction of some new events due to the storage facility associated with aging that isn't in the base facility, there aren't any new events to be introduced, and the consequence of those events we think are enveloped within the existing set of events, which is primarily driven by assembly drops.

DR. HINZE: Do you have any effect of injuries to human entries as a result of underground excavation taken into account, other than radiation? Is this taken into account in the safety analysis?

MR. RICHARDSON: No, we haven't done that type of work yet

MR. HARRINGTON: We will certainly -- sure.

DR. HINZE: Because the difference between the high and the low is quite a difference in terms of length of the excavation, and that's a dicey exercise.

MR. HARRINGTON: Yeah, and consequently or not consequently, but one of the solutions went to smaller capacity waste packages. So if you increase the waste package handling from 11,000 to 18 or so thousand, again, more chances for workers to be injured.

DR. HINZE: That will be taken in, but is this in the ISA?

MR. HARRINGTON: No, it's in the occupational safety discussions. I don't think you do in this one.

DR. HINZE: Yeah, which I believe is somewhat out of my element. It just seemed to me that, you know, there were things other than radiation safety that one should be concerned with, and we see these two operational modes which come up pretty well the same in the TSPA. Is there a difference someplace?

CHAIRMAN HORNBERGER: The difference is that real people get hurt in the occupational safety.

MR. RICHARDSON: I think this would be when Paul mentioned earlier about optimization, I think this would probably be one of the considerations when DOE tries to optimize. I mean, certainly worker safety will be considered.

DR. LEVENSON: It's another one of these group of things that might increase the risk in the preclosure period if you go to low temperature.

DR. HAMDAN: Yeah, I want to go back for a minute to the NRC staff concerns at the exchange, and I take it that these concerns will be taken up in the ISA and some of the license applications. The question I have is are any of these concerns pertinent to the PPSA, and have you done or considered doing an iteration or two to satisfy yourself that these concerns are okay for the site condition?

MR. HARRINGTON: Well, some of the things that came up at the preclosure event or technical exchange would not be captured in the ISA. Probably the area of most significant disagreement continued to be the burn-up credit issue. That's not really an integrated safety assessment issue. We have a topical report and on criticality, and that's where that one is being dealt with.

For those that would be potentially driven by the ISA or incorporated into, that was primarily the tornado and the aircraft, and we had already planned to and are in the process of doing updates to those two sets of analyses, and we agreed with NRC that we would provide them both the methodology for doing the aircraft analysis prior to actually completing it, and then the results of the revision, the update to that aircraft and also to the tornado analysis just to make sure that what we end up with is something that they can accept.

As far as reissuing the PPSA, I don't

think we -- we didn't feel that there was a concern that would causae a site recommendation not to be accepted. I didn't get that sense from NRC. They seemed happy with our proposed resolution and the timing for those.

If I misunderstood their level of concern, now or pretty soon would be a good time to be corrected.

DR. LEVENSON: Any others? Staff? Anyone else have questions?

CHAIRMAN HORNBERGER: Just again to be totally clear, it seems from the earlier presentation that the major thing that you're concerned with in your interaction with NRC to date is this notion of burn-up and how one measures it or accepts the paper record. Is that a fair statement?

You're less worried about the --

MR. HARRINGTON: I think it is.

CHAIRMAN HORNBERGER: Yeah, okay. You're less worried about this aircraft analysis.

MR. HARRINGTON: Right.

CHAIRMAN HORNBERGER: Okay.

MR. HARRINGTON: Right. I really think that we had a very productive technical exchange. I was very pleased with the level of preparation and work that the staff put into it.

I don't think that we had significant issues. We ended up with four agreements in the effectively preclosure area separate from the five that were associated with the waste package and criticality.

Those four were really work activities that we expected to do, and it's not a surprise, not a problem.

CHAIRMAN HORNBERGER: Okay, and you were satisfied at the technical exchange that the communications you had prior to the meeting really led you to communicate effectively at the meeting, and it was a --

MR. HARRINGTON: Very much so.

CHAIRMAN HORNBERGER: Okay, good.

MR. HARRINGTON: I think it's a learning experience for both of us. Some of at least our understanding of the initial staff positions, I think, went beyond what we had done in commercial space for a facility that doesn't have the same risk, and we talked through that and, I think, came to agreement on that.

So, yeah, I was pleased with both the pre-interactions and the tech exchange itself.x

MR. GUNTER: And I just wanted to add that in absence of an IRS audit, the NRC letter that they prepared and sent to us was very beneficial to us. We do appreciate that.

CHAIRMAN HORNBERGER: Okay.

DR. LEVENSON: Okay. It's yours again.

CHAIRMAN HORNBERGER: Thank you very much.

MR. HARRINGTON: You're welcome.

CHAIRMAN HORNBERGER: Very informative presentations.

We are now going to in a moment adjourn. We won't need to be on the record after this. We will reconvene in 15 minutes and work on ACNW reports.

(Whereupon, at 4:14 p.m., the meeting in the above-entitled matter was adjourned.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Nuclear Waste

Docket Number: (Not Applicable)

Location: Rockville, Maryland

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Rebecca Davis

Rebecca Davis
Official Reporter
Neal R. Gross & Co., Inc.

TOTAL SYSTEM PERFORMANCE ASSESSMENT AND INTEGRATION KTI RESOLUTION ACTIVITIES

August 29, 2001

Presented By:
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KEY TECHNICAL ISSUE (KTI): CURRENT STATUS *

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- Multiple Barriers † (Closed Pending)
- Scenario Analysis (Closed Pending)
- Model Abstraction (Open‡)
- Overall Performance Objective† (Closed Pending)

CURRENT STATUS (continued)

FOOTNOTES

* Issue resolution is based on information through the TSPA-SR documents.....does not include:
- Science and Engineering Report (S&ER)
- Supplemental Science and Performance Analyses (SSPA)

† Subissues limited to methodology questions
- Aspects of some analyses that would be included in a potential license application were not reviewed

‡ Model Abstraction remains open due to comments and questions raised through Igneous Activity KTI Technical Exchange

OVERALL APPROACH/METHOD FOR TECHNICAL EXCHANGES

- Focus on specific NRC staff concerns or questions
- DOE presents the status of activities and path forward items related to each KTI subissue
- NRC staff identifies what additional information is needed from DOE
- Formal agreement is reached on:
  - What additional information is to be provided
  - Schedule for submittal
  - Agreement documented in meeting summary

RELATIONSHIP BETWEEN PROCESS-KTIs AND TSPAI KTI

- TSPAI Subissues overlap with process-KTI Subissues

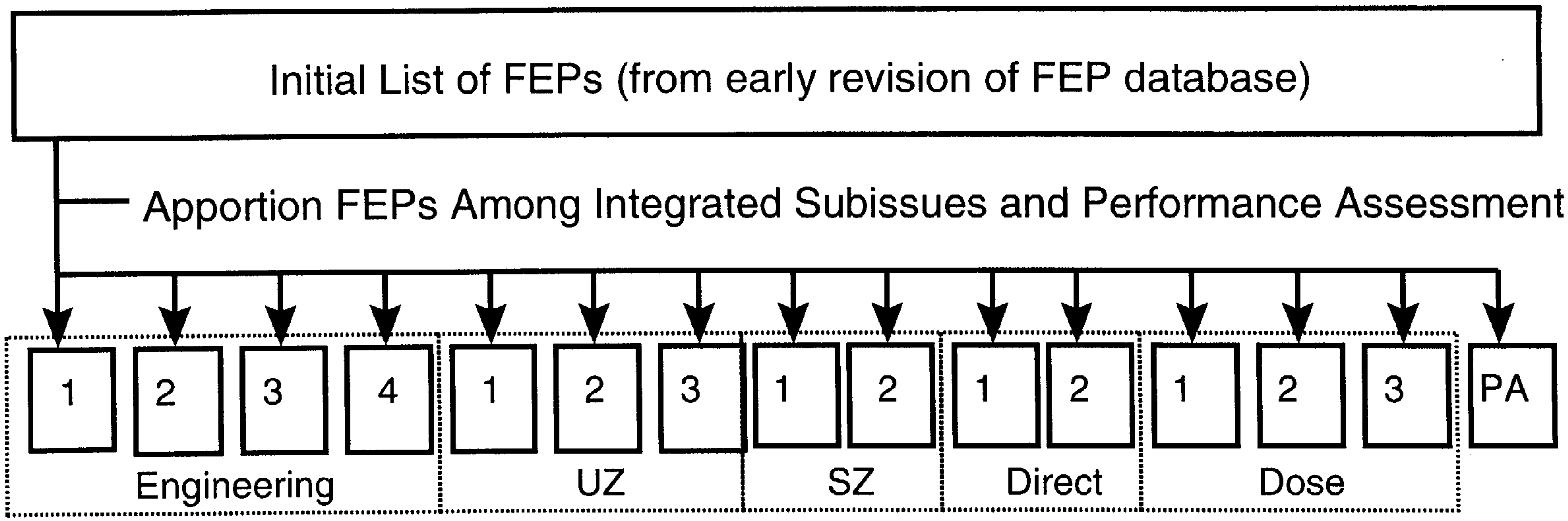
- TSPAI Subissues establish a framework for evaluating DOE's performance assessment (PA)
  - Integrated Subissues (ISIs) were developed for reviewing DOE model abstractions
  - ISIs include process-KTI subissues
  - ISI teams involve multiple KTI teams

COORDINATION AMONG KTIs

- Identified roles and responsibilities among KTIs for different sections of TSPAI Subissues
  - Process-KTIs have the lead for parts of Model Abstraction and Scenario Analysis Subissues
    - Features, Events, and Processes (FEPs) within their respective areas
    - Basis for events with probability greater than 10^{-8}/year
    - Four of five acceptance criteria within Model Abstraction

- Integration was performed through ISI teams and through integration at the subissue level

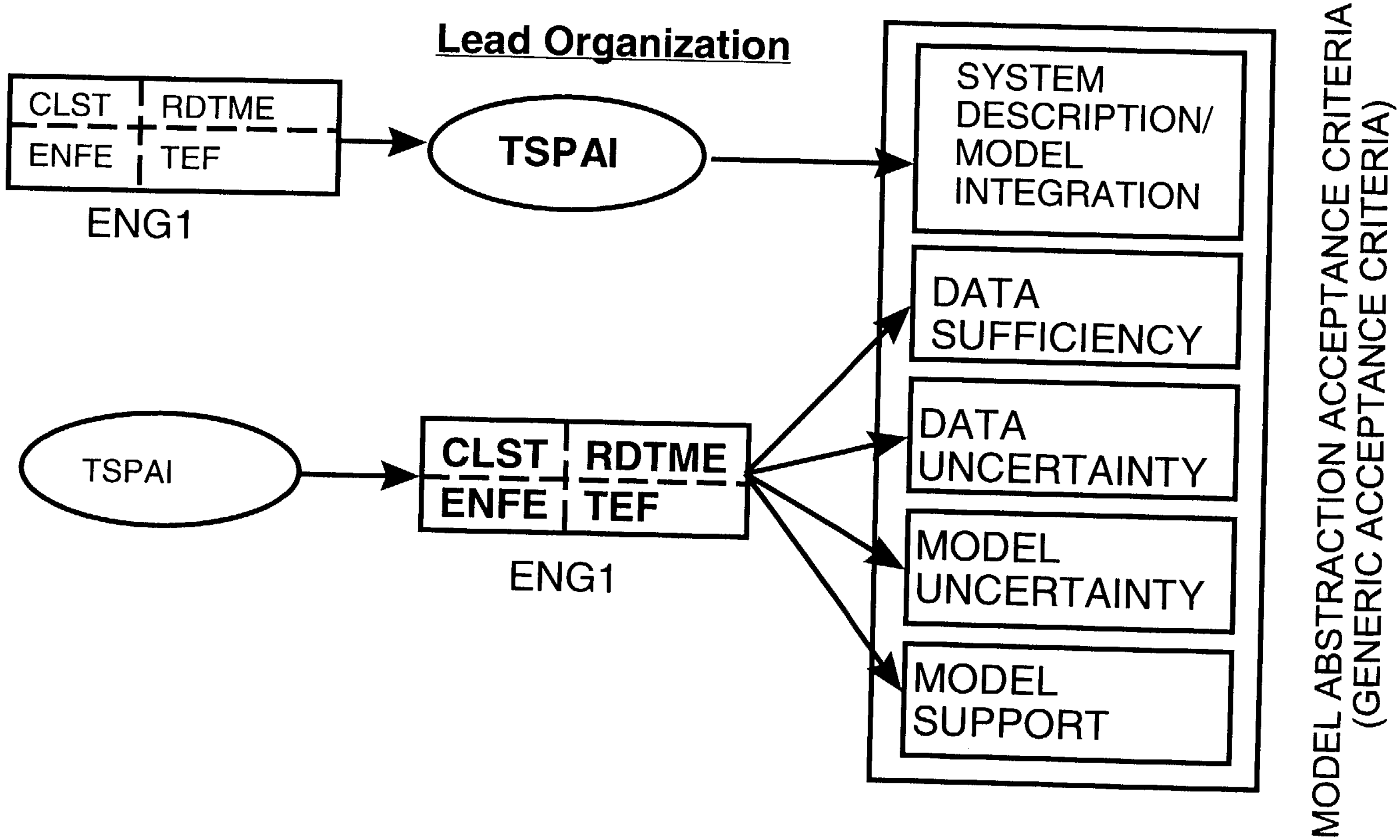
SCENARIO ANALYSIS



- Integrated subissue teams
  - Evaluated list for comprehensiveness
  - Evaluated screening arguments
- TSPAI KTI
  - Evaluated "system-level" FEPs
  - Coordinate review (integration, consistency)
  - Methodology

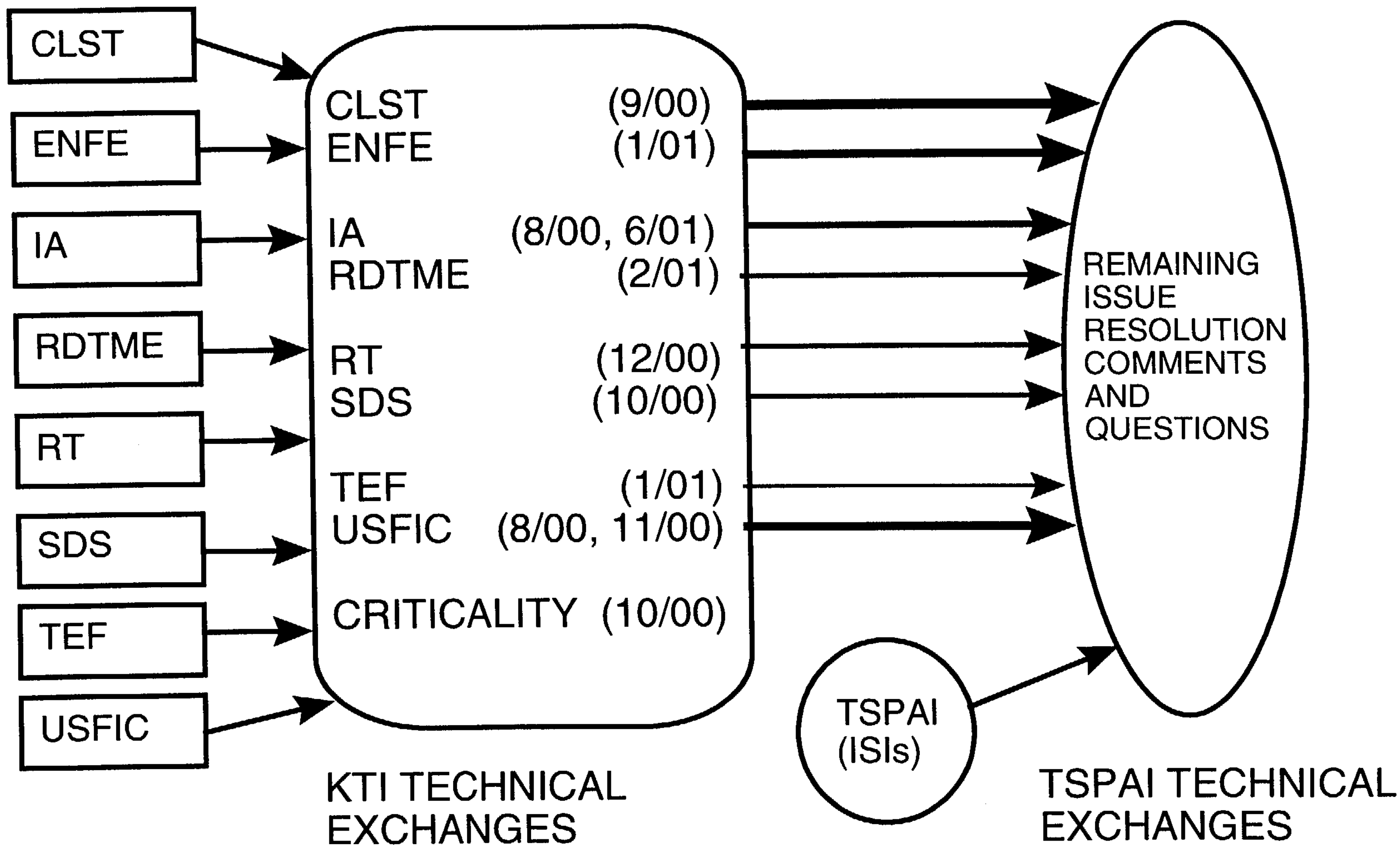
MODEL ABSTRACTION

Degradation of Engineered Barriers (ENG1) Example



MODEL ABSTRACTION

APPROACH TO ISSUE RESOLUTION



COMMENT SUMMARY

- Scenario Analysis
  - 7 General questions or comments
  - 103 Comments on specific FEPs
- Multiple Barriers
  - 11 Comments
- Overall Performance Objective
  - 2 General comments (with examples)
  - 27 Specific comments

COMMENT SUMMARY (continued)

- Model Abstraction
  - 4 General comments with examples
  - 4 General transparency/traceability comments
  - 112 Specific comments
    - Engineering ISIs (43)
    - Dose ISIs (43†)
    - Unsaturated zone ISIs (17)
    - Direct ISIs (6†)
    - Saturated zone (SZ) ISIs (3)

† Other comments are being addressed through Igneous Activity Technical Exchanges

TOP ISSUES

- Scenario Analysis
  - Comprehensive List of FEPs

- Multiple Barriers
  - Approach to Multiple Barriers

- Model Abstraction
  - Data uncertainty in corrosion
  - Quantity and chemistry of water contacting waste package and drip shield
  - Approach to developing abstractions and parameter value distributions

- Overall Performance Objective
  - Stability of results
  - Model validation and computer code verification (also applies to Model Abstraction)

DOE SCENARIO SCREENING METHODOLOGY: NRC STAFF VIEWS

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BACKGROUND: REGULATORY FRAMEWORK

Scenario Analysis

- Proposed Part 63 requires a systematic analysis of FEPs that might potentially affect geologic repository performance
  - Specifies consideration of those FEPS that could materially affect compliance with the overall system performance objective or have potentially adverse effects on performance
  - Events with annual probabilities less than 10^{-8} can be excluded
  - Rule does not specify the manner by which FEPs should be investigated

- Purpose of FEPs screening process
  - What has been considered, and
  - What has been excluded from any PA calculation

NRC EXPECTATIONS

Scenario Analysis

- DOE can screen FEPs using whatever method(s) it chooses
- In general, method should include:
  - Adequate technical justification
  - Well-documented
- In particular, method should:
  - Be thorough
  - Provide correct characterization and treatment of FEPs as singular or universal
  - Have sound probabilistic arithmetic
  - Consider FEP representativeness and variability
- Attributes of an acceptable approach
  - Identification of initial list of FEPS
  - Categorization of FEPS
  - Screening of the initial list of FEPS
  - Formation of scenario classes
  - Screening of scenario classes

DOE APPROACH

Scenario Analysis

- Foundation: Nuclear Energy Agency FEP database

- Additional FEPs identified though:
  - DOE Site characterization activities
  - Independent NRC staff review comments

- Results:
  - 1808 total FEP entries
  - 328 individual FEPS
  - 135 FEP classes

- FEPS documented in Analysis Model Reports (6)

CONCLUSIONS/PATH FORWARD

Scenario Analysis

- DOE FEP evaluation may be comprehensive ... but process is not auditable
  - Identification and categorization steps combined
  - Level of detail not always uniform
  - Transparency and traceability relative to larger PA process needs to be improved

- To address staff concerns, DOE agreed to the following process enhancements (commitments)
  - Revise FEP descriptions and level of detail (transparency)
  - Improve documentation of FEP disposition in overall TSPA process (traceability)
  - Development of electronic FEP database to permit configuration management of FEP process

DOE APPROACH TO THE TREATMENT OF MULTIPLE BARRIERS

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DOE APPROACH

Multiple Barriers

- Documented preliminary approach in the Repository Safety Strategy (Version 4) and TSPA-SR

- Identified natural and engineering barriers

- Performed TSPA model simulations of barriers

- Presented dose curves for radionuclides

NRC CONCERNS

Multiple Barriers

- Reviewed preliminary approach and provided 11 comments to DOE
- Documentation insufficient to understand approach and results for demonstrating barrier's capabilities
- Variability of natural barriers and uncertainty in natural and engineered barrier performance needs analysis and documentation

RESOLUTION/PATH FORWARD

Multiple Barriers

- Two agreements reached at Technical Exchange:
  - Provide descriptive documentation considering parameter and model uncertainty
  - Document barrier capabilities considering variability, independence and interdependence of barrier functions
- DOE will document approaches in FY02 TSPA-Methods and Assumptions Report and present results in any potential License Application

DOE APPROACH TO MODEL ABSTRACTION

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DOE APPROACH

Model Abstraction: Uncertainty in Corrosion

- Complete empirical testing of general corrosion in 4 primary environments
- Utilize the empirical test results to define the distribution of general corrosion rates for utilization in PA
- Assert that the environments are aggressive and bounding, therefore general corrosion rates are conservative
- State that the data used demonstrate general corrosion rates are decreasing in time, therefore use of 2-year test results are conservative
- Utilize Gaussian Variance Partitioning (GVP) in the PA to represent the impact of uncertainty and variability in the general corrosion data

NRC CONCERNS

Model Abstraction: Uncertainty in Corrosion

- Measurement techniques used do not have enough resolution to characterize general corrosion rates
- Presence of silica in all testing solutions confounds test interpretation and uncertainty representation
- Crevice geometry weight loss samples show statistically different (higher) rates than non-crevice weight loss geometry samples (higher rates for the crevice area, how much higher ?)
- True variability of the general corrosion rates is undefined when weight loss (corrosion) and weight gain (silica precipitation) are competing
- Alternative explanation of the temporal decrease in rates are possible (silica precipitation and re-deposition of corrosion products reduce the reactive surface area on samples)

NRC CONCERNS (continued)

Model Abstraction: Uncertainty in Corrosion

- Uncertainty resulting from utilizing a limited sample size to define silica precipitation correction and high corrosion rates was not represented in the PA for TSPA-SR (or the SSPA)

- If a large portion of the variance in the general corrosion rates is due to measurement uncertainty, representation of epistemic uncertainty in the TSPA-SR may result in risk dilution

- DOE's representation of uncertainty and variability in the TSPA-SR via utilization of GVP lacked transparency*

* 100 percent uncertainty case and 100 percent variability case showed little sensitivity. Independent modeling by NRC suggests potentially large influence, dependent on release rates from the engineered barrier system. Still unclear whether the DOE/NRC were comparing apples and apples.

RESOLUTION/PATH FORWARD

Model Abstraction: Uncertainty in Corrosion

- CLST agreements reached (not inclusive):
  - Addressing the collection of general corrosion information with techniques that have better resolution, and
  - Resolving the influence of silica on the test results

- TSPAI reached 3 agreements addressing uncertainty in corrosion rates with an emphasis on representation of all important sources of uncertainty in future PA

- CLST, ENFE, and TSPAI agreements demonstrate an integrated review of the technical issues

- Integrated review covered both fundamental information needs and treatment/representation of uncertainty in PA

DOE APPROACH

Model Abstraction: Quantity and Chemistry of Water Contacting the WP/DS

- DOE has completed testing of WP/DS materials in solutions that they assert are bounding and conservative

- DOE has defined environmental conditions based on coupled thermo-hydro-chemical (THC) modeling for seepage compositions that are then concentrated as a result of evaporative processes

- Evaporative processes addressed through modeling and experimentation

- DOE modeling for the TSPA-SR has suggested evolution of solutions to a sodium bicarbonate type brine

NRC CONCERNS

Model Abstraction: Quantity and Chemistry of Water Contacting the WP/DS

- DOE has not demonstrated that all expected solution compositions will evolve into sodium bicarbonate type brine, therefore resulting in concern about the assertion of bounding environments for corrosion testing

- DOE has not completed an integrated analysis of important sources of uncertainty for the THC models and in-drift environmental conditions, and therefore have not represented this uncertainty in the PA

- The THC seepage model propagated very limited uncertainty into downstream models for TSPA-SR

- Various geochemical modeling activities did not appropriately address/represent uncertainty in fundamental thermodynamic information

RESOLUTION/PATH FORWARD

Model Abstraction: Quantity and Chemistry of Water Contacting the WP/DS

- ENFE agreements reached (not inclusive):
  - Address the characterization of uncertainty/variability for the THC seepage model
  - Evaluate the uncertainty in in-drift geochemical models

- TSPAI reached 5 agreements addressing uncertainty, integration, and model confidence for the prediction of in-drift geochemical environments and their abstraction in the PA

- CLST, ENFE, and TSPAI agreements demonstrate an integrated review of the technical issues

- Integrated review covered both fundamental information needs and treatment/representation of uncertainty in PA

DOE APPROACH

Model Abstraction: Approach to Developing Abstractions and Parameter Value Distributions

- DOE has elected to use professional judgement and (selected) quantification for the simplifications utilized for model abstractions
- Parameter distributions are selected by process-level analysts (or TSPA personnel as appropriate)
- 'Local' arguments are typically made for simplifications
- 'Local' arguments are typically made for conservative selections
- Comparisons of parameter distributions selected to the information available to support their selection was typically not given in the TSPA-SR

NRC CONCERNS

Model Abstraction: Approach to Developing Abstractions and Parameter Value Distributions

- While NRC acknowledges that it is intractable to represent all uncertainty and variability in a PA, care must be taken that important uncertainty and variability are not eliminated in the PA (or unevaluated)
- Simplification of abstractions can and should be used, but the simplification process should be systematic and appropriately address uncertainties
- If conservatism is utilized to substitute for lack of information, the selection of the conservative model or data distribution should be systematic and appropriately address uncertainties
- Typically, comparison of abstractions utilized in the PA to process model output are not documented
- Typically, comparisons of data distributions to the data used to develop those distributions are not provided in the TSPA

RESOLUTION/PATH FORWARD

Model Abstraction: Approach to Developing Abstractions and Parameter Value Distributions

- Four agreements were reached that will address concerns regarding model abstraction simplifications, representation of data uncertainty, consistency of the propagation of models from supporting documentation to the TSPA, and utilization of conservatism

- Agreements should ensure more consistency in the TSPA process

- Comparison of abstractions to the process-model output and the comparisons of the data distributions to the supporting data should help develop confidence in both the abstraction process and the representation of data uncertainty in the TSPA

OVERALL PERFORMANCE OBJECTIVE: STABILITY OF RESULTS

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DOE APPROACH

Overall Performance Objective: Stability of Results

- Statistical stability of mean dose versus time curve is evaluated by visual comparison of plots for different numbers of realizations

- Stability of TSPA model abstractions with respect to variations in the degree of spatial and temporal discretization has not been tested by DOE:
  - **Spatial**: Number of infiltration bins, number of thermohydrology bins
  - **Temporal**: Time step size for seepage, corrosion, spent fuel dissolution, radionuclide transport; number of (future) climate states

NRC CONCERNS

Overall Performance Objective: Stability of Results

- Demonstration of convergence of the LHS method as applied in the TSPA-SR requires greater mathematical rigor

- Submodel results such as BCDFs and SZ transport kernels are developed using a limited number of realizations

- Because the mean is highly affected by extreme values, and the likelihood of observing an extreme condition is often increased as spatial resolution is increased, DOE should demonstrate that the peak mean dose does not increase as spatial resolution is increased

- DOE has not demonstrated that use of finer temporal discretization in the TSPA-SR code does not change the peak mean dose

RESOLUTION/PATH FORWARD

Overall Performance Objective: Stability of Results

- DOE will describe the statistical measures to be used in the method they will use to demonstrate the statistical convergence of the LHS method in computing the mean dose versus time curves
- DOE will conduct appropriate analyses and provide documentation that demonstrates that the results of the PA are stable with respect to spatial and temporal discretization

OVERALL PERFORMANCE OBJECTIVE: MODEL VALIDATION AND COMPUTER CODE VERIFICATION

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BACKGROUND

Overall Performance Objective: Model Validation and Computer Code Verification

- Regulatory requirements
  - Test/corroborate confidence in PA models
  - Provide technical basis for PA models
  - Confirm that barrier behavior is consistent with PA assumptions

- Traditional "scientific model validation" not sought for compliance demonstration

- Working Definitions of "Confidence Building"
  - **Software Confidence**: Provides assurance that a computer code correctly performs the operations in a numerical model
  - **Model Confidence**: Provides assurance that a model, as embodied in a computer code, is a correct representation of process or system

- NUREG-1636/SKI Report 99:2 – The White Paper
  - Example approach to model validation
  - Not intended as a *defacto* staff position

DOE APPROACH

Overall Performance Objective: Model Validation and Computer Code Verification

- Elements of software confidence building is TSPA-SR
  - Verification limited to component models
  - Limited set of hand calculations

- Model confidence building
  - Documented in development of S&ER and AMRs
  - Quality assurance (QA) requirements specified in AP3.10Q
    - States that peer review is an equal alternative to experiments

- DOE Corrective Action Reports (CARs)
  - CAR BSC-01-C-001: Failure to consistently implement QA requirements of AP3.10Q
  - CAR YMSCO-01-C-002: (1) Failure to implement QA program related to software; (2) Lack of effective independent validation and verification

NRC CONCERNS

Overall Performance Objective: Model Validation and Computer Code Verification

- Not apparent that DOE has a vision, philosophy, or strategy for confidence building

- Software confidence concerns:
  - No DOE verification plan
  - Errors found in hand calculations: models used outside the range; significance of warnings/error not explained
  - TSPA model correctness not verified for input value extremes
  - No verification at the TSPA code level

NRC CONCERNS (continued)

Overall Performance Objective: Model Validation and Computer Code Verification

- Model confidence concerns:
  - DOE model validation requirements not consistently implemented
  - Submodel validation used as a surrogate for system-level validation
  - Conceptual models validated but not the corresponding mathematical models (e.g., biosphere)
  - No objective comparisons for component models
  - Peer review appears to have been used as a substitute for objective information that is reasonably available

RESOLUTION/PATH FORWARD

Overall Performance Objective: Model Validation and Computer Code Verification

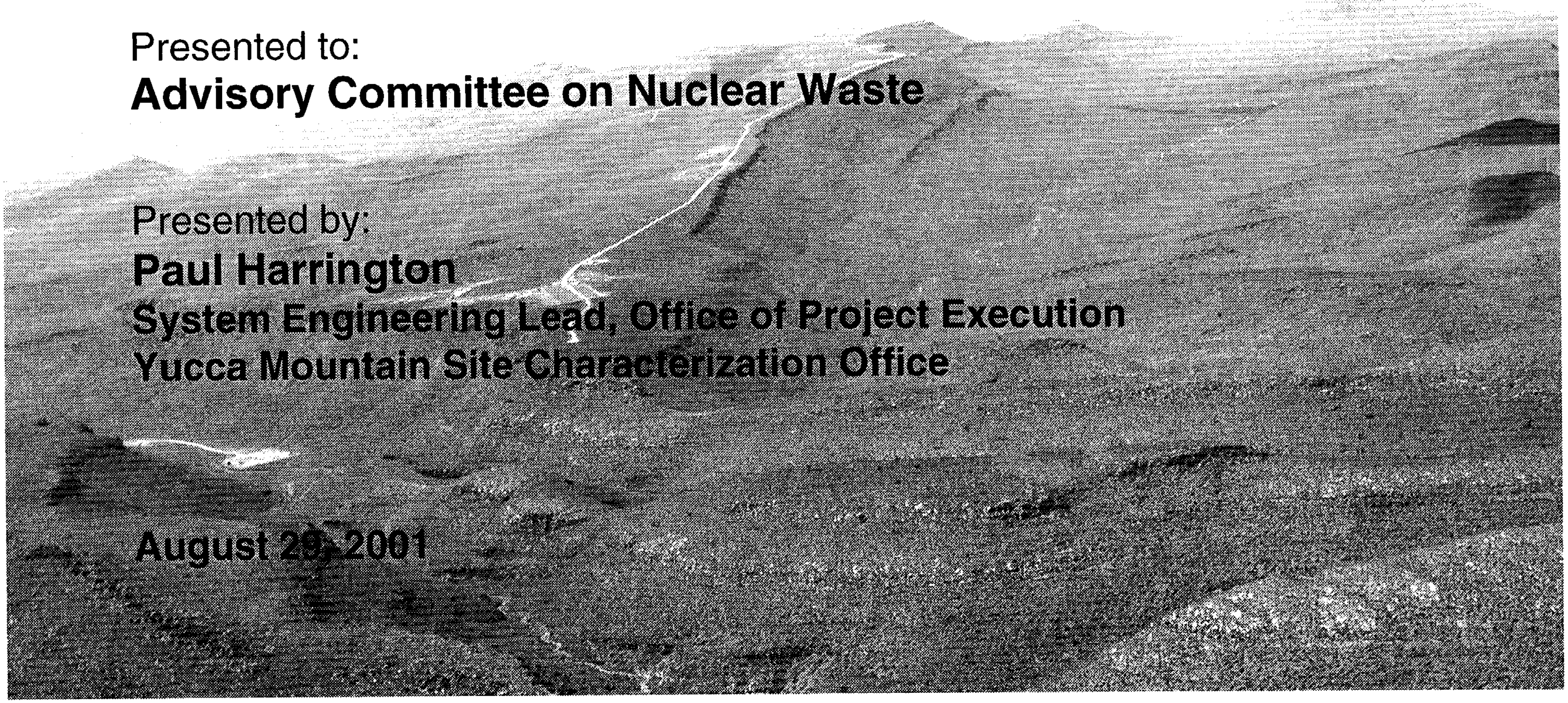
- Software confidence building ... DOE to provide
  - An overall strategy/plan for PA software verification
  - Describe minimum requirements for completing software verification of existing computer codes
  - Describe how software verification actions and results will be documented

- Model confidence building ... DOE to provide
  - An overall strategy/plan for PA model validation
    - Justification for the selection of the validation criteria (qualitative and/or quantitative)
    - Consistent with existing DOE QA requirements/directives
    - Once implemented, document statements of model validity results against validation criteria

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

YUCCA MOUNTAIN PROJECT

DOE-Yucca Mountain Preclosure Plans



Presented to:
Advisory Committee on Nuclear Waste

Presented by:
Paul Harrington
System Engineering Lead, Office of Project Execution
Yucca Mountain Site Characterization Office

August 29, 2001

Objective

- Describe preliminary preclosure safety assessment for site recommendation
- Describe integrated safety analysis for the proposed license application
- Provide overview of recent engineering developments

Preliminary Preclosure Safety Assessment

- **Purpose**
  - Support site recommendation suitability evaluation
  - Document preclosure safety assessment performed in FY 2000, updated for repository flexible design features in FY 2001

Preliminary Preclosure Safety Assessment*
(Continued)

- **Scope**
  - Identification of applicable external and internal hazards
  - Analysis of design basis event frequencies and consequences
  - Identification of structures, systems, and components important to safety
  - Strategies for criticality safety, fire protection, radiation protection, and management of low-level wastes

* *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation*, TDR-MGR-SE-000009 Rev 00 ICN 03

External Events Hazard Analysis

- Generic checklist of 53 event categories
- Screened for
  - Potential at site
  - Consequences of process significant in preclosure period
  - Event frequency ($\geq 1x10^{-6}$/yr)
- Screening and grouping reduced number of events incorporated into design basis: loss of site power; seismic vibrations/displacement; flood; winds/tornadoes, and missiles

Internal Event Hazard Analysis

- Generic list of 6 categories of hazards (energy forms) that could potentially impact waste form and cause release of radioactivity
  - Collision/crushing
  - Chemical/flooding
  - Explosion/implosion
  - Fire
  - Radiation/fissile
  - Thermal

Internal Event Hazard Analysis
(Continued)

- Systematic analysis of each operational area for presence of each hazard and potential interactions with waste form
- Preliminary hazards list used as initiating events input to event sequence/design basis event analyses

Proposed 10 CFR 63 Design Basis Event Categorization

- **Category 1**
  - **Proposed 10 CFR 63.2 definition**

    "Those natural and human-induced events that are expected to occur one or more times before permanent closure of the GROA"
  - **Interpreted as those conditions of normal operation that are expected to occur one or more times during preclosure facility lifetime (i.e., taken to mean events greater than or equal to 10^{-2} per year based on a 100-year preclosure period)**

Legend
GROA - Geologic Repository Operations Area

Proposed 10 CFR 63 Design Basis Event Categorization

(Continued)

- **Category 2**
  - Proposed 10 CFR 63.2 definition

    "Other natural and man-induced events that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository"
  - Interpreted as design basis events occurring with frequencies ranging from Category 1 to 10^{-6}/yr (i.e., Category 1 > $F_i \geq 10^{-6}$/yr)

Analyses to Identify Design Basis Events

- Developed event trees to identify
  - Event sequences that could lead to release
  - Prevention and mitigation functions of structures, systems, and components
- Estimated frequencies and probabilities using
  - Industry data, Licensing Event Reports, generic databases
  - Maximum throughput rates for operations
  - Fault tree analysis (e.g., heating ventilation and air conditioning; equipment drop)
- Binned event sequences according to
  - Design basis event frequency Category 1 or 2; or beyond design basis events

Category 1 Internal Design Basis Event Sequences

Event Sequence #	Event Description	Location	Frequency (per Year)
1-01	Spent nuclear fuel assembly drop onto another spent nuclear fuel assembly in cask	Assembly transfer system pool	0.2
1-02	Spent nuclear fuel assembly collision with pool wall	Assembly transfer system pool	0.04
1-03	Spent nuclear fuel assembly drop onto empty basket	Assembly transfer system pool	0.04
1-04	Spent nuclear fuel assembly drop onto another spent nuclear fuel assembly in basket staging rack (lowering into)	Assembly transfer system pool	0.2
1-05	Basket drop onto another basket in basket staging rack (lifting out)	Assembly transfer system pool	0.04
1-06	Basket drop onto another basket in pool (transfer into pool storage)	Assembly transfer system pool	0.04
1-07	Basket drop onto another basket in pool (transfer out of pool storage)	Assembly transfer system pool	0.04
1-08	Basket drop onto transfer cart or pool floor	Assembly transfer system pool	0.04
1-09	Basket drop back into pool	Assembly transfer system pool	0.04
1-10	Basket drop onto assembly transfer system cell floor	Assembly transfer system cell	0.04
1-11	Basket drop onto another basket in dryer	Assembly transfer system cell	0.04
1-12	Spent nuclear fuel assembly drop onto another spent nuclear fuel assembly in dryer	Assembly transfer system cell	0.2

Summary of Preclosure Category 1 Design Basis Event Doses for the Public and Workers

Case	Dose Type	Radiation Dose	Dose Standard
Offsite Public, Category 1 DBE (including normal operational releases)	Annual total effective dose equivalent	0.06 mrem/yr	10 mrem/yr (10 CFR 20) 15 mrem/yr (40 CFR 197) 25 mrem/yr (proposed 10 CFR 63) 100 mrem/yr (10 CFR 20)
	External exposure	<< 2 mrem/hr	2 mrem in any 1 hour/yr (10 CFR 20)
Workers, Category 1 DBE (including normal operational effluents and emissions)	Annual total effective dose equivalent	0.01 rem/yr	5 rem/yr (proposed 10 CFR 63)
	Organ or tissue plus deep dose	0.1 rem/yr	50 rem/yr (10 CFR 20)
	Skin and extremities	0.1 rem/yr	50 rem/yr (10 CFR 20)
	Lens of eye	0.1 rem/yr	15 rem/yr (10 CFR 20)

Summary of Preclosure Category 2 Design Basis Event Offsite Doses

Event Sequence Number	Event Description	Skin Dose (rem)	Organ or Tissue Dose (rem)	Lens of the Eye Dose (rem)	Total Effective Dose Equivalent (rem)
2-01	Spent nuclear fuel assembly basket collision during transfer	0.007	0.002	0.007	0.0004
2-02	Uncontrolled descent of incline transfer cart	0.007	0.002	0.007	0.0004
2-03	Handling equipment drop onto spent nuclear fuel assembly basket in pool	0.007	0.002	0.007	0.0004
2-04	Handling equipment drop onto spent nuclear fuel assembly basket in cell	0.005	0.01	0.008	0.003
2-05	Unsealed disposal container collision	0.03	0.07	0.05	0.02
2-06	Unsealed disposal container drop and slapdown	0.03	0.07	0.05	0.02
2-07	Handling equipment drop onto unsealed disposal container	0.03	0.07	0.05	0.02
2-08	Unsealed transportation cask drop into cask preparation pit	0.04	0.1	0.06	0.02
2-09	Unsealed transportation cask drop into cask unloading pool	0.04	0.01	0.04	0.002
Proposed 10 CFR 63.111 (64 FR 8640) Dose Standard		50 rem	50 rem	15 rem	5 rem

NOTE: The radiation dose for the lens of the eye is the sum of the skin dose and the total effective dose equivalent.

Preclosure Safety Strategies for Design of Monitored Geologic Repository Operations

- Preclosure safety strategy establishes site recommendation design requirements
  - Ensures compliance with proposed 10 CFR 63
  - Accommodates constraints of postclosure safety case
  - Minimizes design complexity
- Basic strategies defined for each preclosure operation
  - Containment/confinement augmented by prevention (i.e., offsite dose < 25 mrem/yr for Category 1; < 5 rem for Category 2)
  - Prevention augmented by containment/confinement (i.e., frequency < 1×10^{-6}/yr to screen out or < 1×10^{-2}/yr not expected in preclosure time period)
- Design criteria to implement strategies incorporated into system description documents

Additional Safety Strategies for Design Considerations

- Preliminary Preclosure Safety Assessment provides overviews of preclosure safety strategies
  - Criticality safety
  - Radiation protection including as low as is reasonably achievable
  - Fire protection
  - Radioactive/hazardous waste management
- Strategies include
  - Identification of operations where applicable
  - Identification of precedent regulations, guidelines, and standards
  - Approaches for applying analyses, design features, and administrative controls

Preliminary Preclosure Safety Assessment Considered Lower Temperature Design Parameters

- **Addressed affected parameters such as waste package size, spacing, excavated drift length, forced/natural ventilation, surface aging concept, and various preclosure periods**
- **Concluded**
  - **Lower-temperature mode, alone, introduces no new bounding design basis event sequences**
  - **Lower-temperature mode with extended aging surface facility may introduce new design basis events; none expected to exceed prior set identified for higher temperature modes**

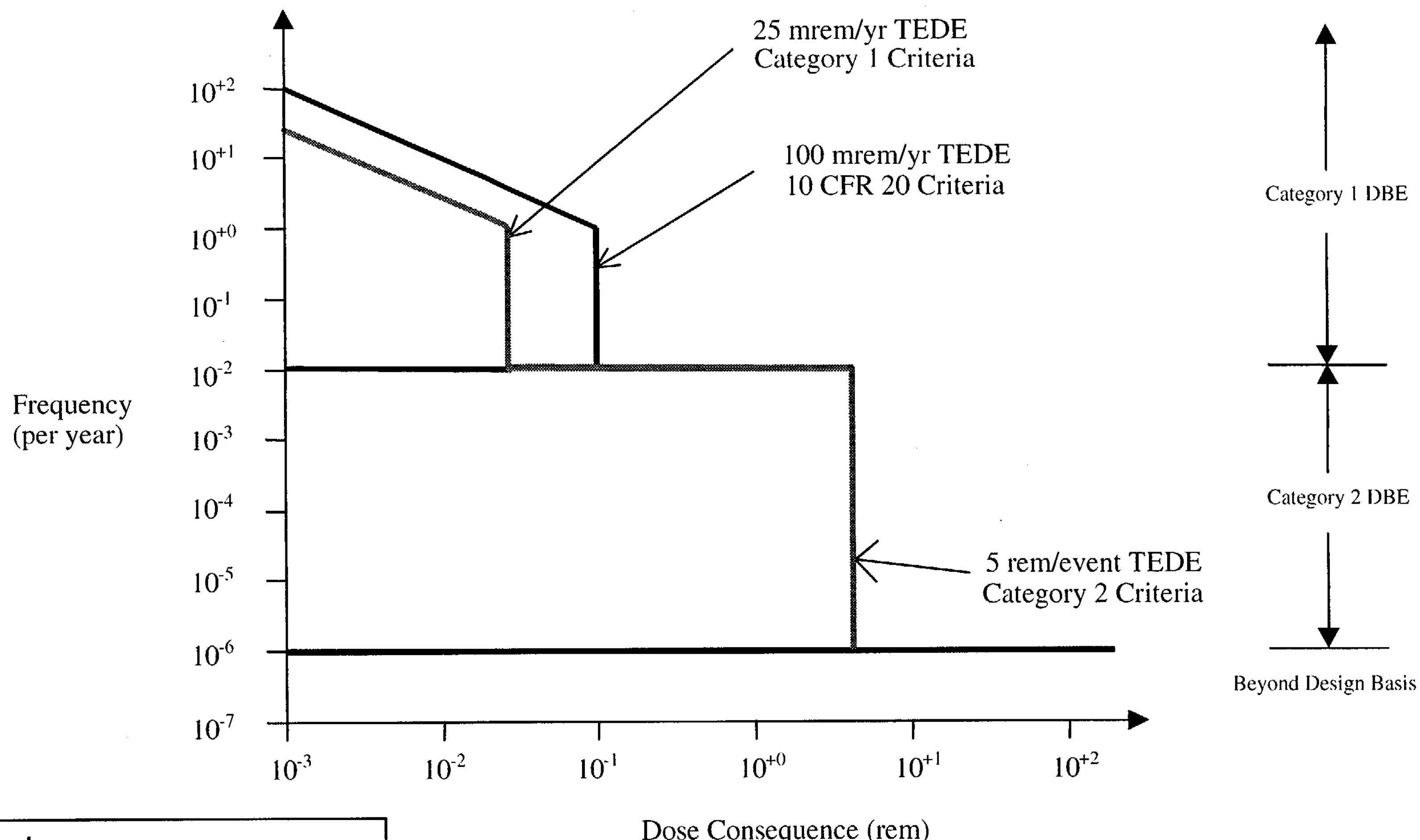
Summary
Preliminary Preclosure Safety Assessment

- Preliminary Preclosure Safety Assessment was performed to support site recommendation
- Approach was essentially same as process for integrated safety analysis to be performed in support of license application but much less detailed
- Offsite and worker doses demonstrated to be well within regulatory limits of proposed 10 CFR 63
- Preclosure safety strategies were developed, incorporated in analyses, and implemented in design criteria for structures, systems, and components important to safety
- Results are not significantly affected by flexible repository design alternatives

Integrated Safety Analysis

- Describe how an integrated safety analysis will be developed to support a potential license application for a repository
  - How integrated safety analysis process is risk-informed
  - How integrated safety analysis will take advantage of industry precedent
  - Discuss the products that will support the integrated safety analysis

Interpretation of Offsite Dose Criteria for Proposed 10 CFR 63 Risk Criteria Diagram



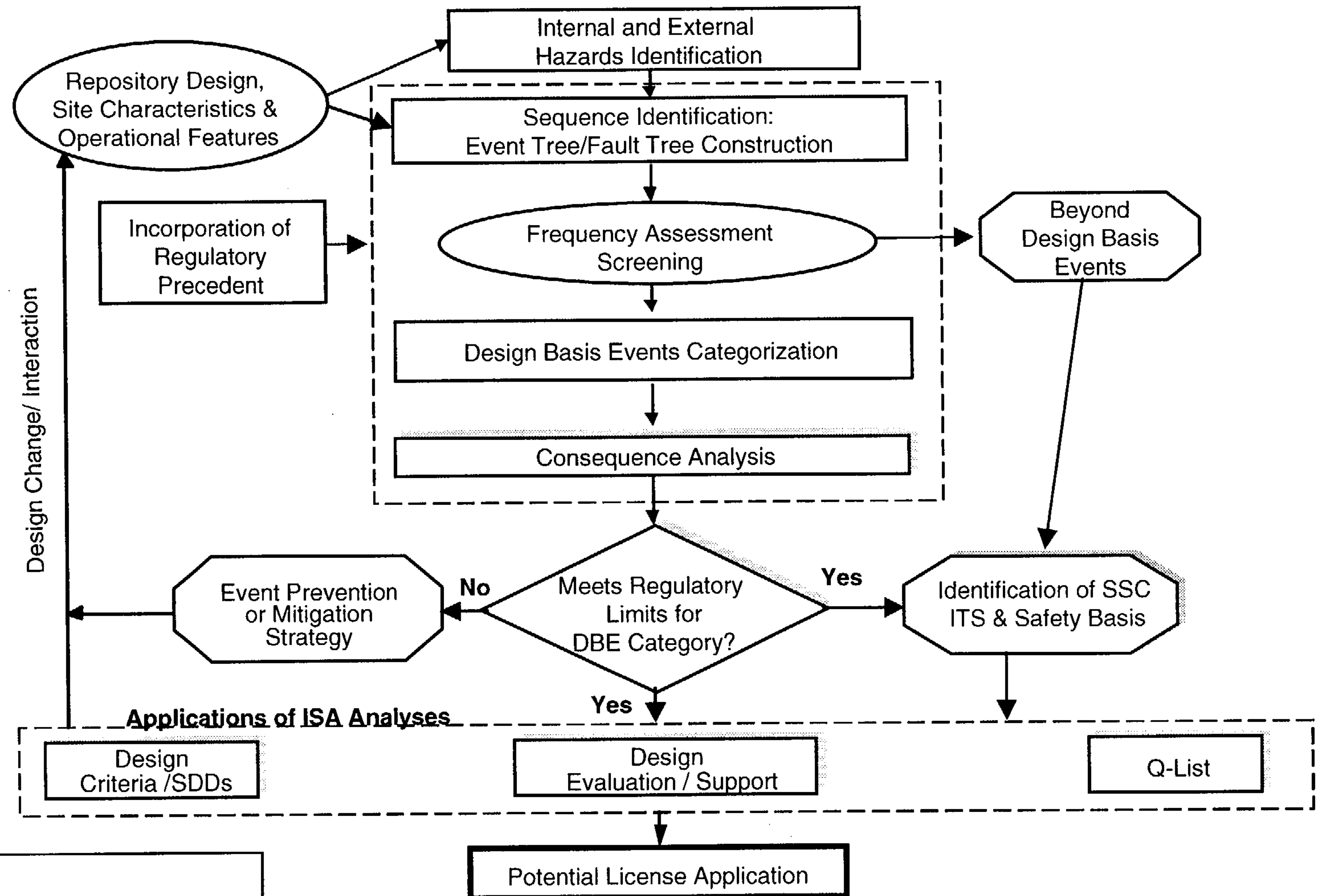
Legend

TEDE - Total effective dose equivalent
DBE - Design basis event
mrem - millirem
rem - measure of absorbed dose

What is an Integrated Safety Analysis?

- **Proposed 10 CFR 63.2**
  - "Integrated safety analysis means an analysis to identify hazards and their potential for initiating event sequences, the potential event sequences, and the site, structures, systems, components, equipment, and activities of personnel, that are relied on for safety. As used here, integrated means joint consideration of safety measures that otherwise might conflict, including, but not limited to, integration of fire protection, radiation safety, criticality safety, and chemical safety measures"

Overview of the Integrated Safety Analysis Process



Legend
DBE - Design basis event
ISA - Integrated safety analysis
ITS - Important to safety
SDD - System description document
SSC - Structure, system, and component

Integrated Safety Analysis Products

Hazards Identification → **DBE Categorization** → **Consequence Analysis** → **Identification of SSCs and Controls to Prevent and/or Mitigate** → **Integrated Safety Analysis**

Hazards Identification

MGR External Hazards (ANL-MGR-SE-000004)

MGR Internal Hazards (ANL-MGR-SE-000003)

Surface Fire Hazards Analysis

Subsurface Fire Hazards Analysis

DBE Categorization

Categorization of Design Basis Events

Aircraft Hazards (ANL-WHS-SE-000001)

Wind/Tornado (ANL-MGR-SE-000001)

Industrial/Military (ANL-WHS-SE-000004)

Rainstorm/Flooding

Seismic

DBE Fire

Loss of Offsite Power

HVAC Fault Tree Analysis

Waste Handling Fault Tree Analysis

Component Failure/Reliability Analysis Database

Preclosure DBEs Related to WP (ANL-MGR-MD-000012)

Preclosure Criticality Evaluations

Transporter Reliability Analysis

WP DBE Structural Analysis (Drop Height)

Shipping Cask Structural Analysis (Drop Height)

Rockfall/Handling Events

Consequence Analysis

Potential Consequences of Design Basis Events

Potential Consequences of Selected Beyond Design Basis Events

Commercial SNF Release Fractions (ANL-WHS-SE-000002)

Cat 1 DBE Worker Dose Evaluations

ALARA Evaluations

Radiation Protection Program

Atmospheric Dispersion Factors

Preclosure SNF Source Terms

Surface Normal Releases

Subsurface Normal Releases

Naval Fuel DBE Source Terms/Releases

DOE Fuel Groups/Source Terms

DOE Canister Drop Tests

Pu Can-In-Canister Structural Analysis

Identification of SSCs and Controls to Prevent and/or Mitigate

Nuclear Safety Design Basis Report

Classification Analyses

Q-List (YMP/90-55Q)

Classification Procedure QAP-2-3 (AP-2.22Q)

Grading Process/Procedure

Design Criteria Document

System Description Documents

Integrated Safety Analysis

Integrated Safety Analysis

ISA Guide

Hazards Procedure

ISA Procedure

Legend

ALARA	- As low as is reasonably achievable
DBE	- Design basis event
HVAC	- Heating, ventilation and air conditioning
MGR	- Monitored geologic repository
SNF	- Spent nuclear fuel
SCCs	- Structure, System, and Component
ISA	- Integrated safety analysis
WP	- Waste package

Guidance from Current Regulatory Developments and Precedents

- Following licensee and NRC activities for facilities such as
  - Mixed Oxide Fuel Fabrication (10 CFR 70)
  - Private Fuel Storage Facility (10 CFR 72)
  - Pebble-Bed Pre-Application (10 CFR 50)
  - Navy and DOE Spent-Fuel Facilities (Idaho National Engineering and Environmental Laboratory)
  - Decommissioning of spent fuel storage pools (NUREG-1726)
- Standard Review Plans (e.g., NUREG-1536, -1567, -1718)
- Integrated Safety Analysis guidance (e.g., NUREG-1520, -1513)

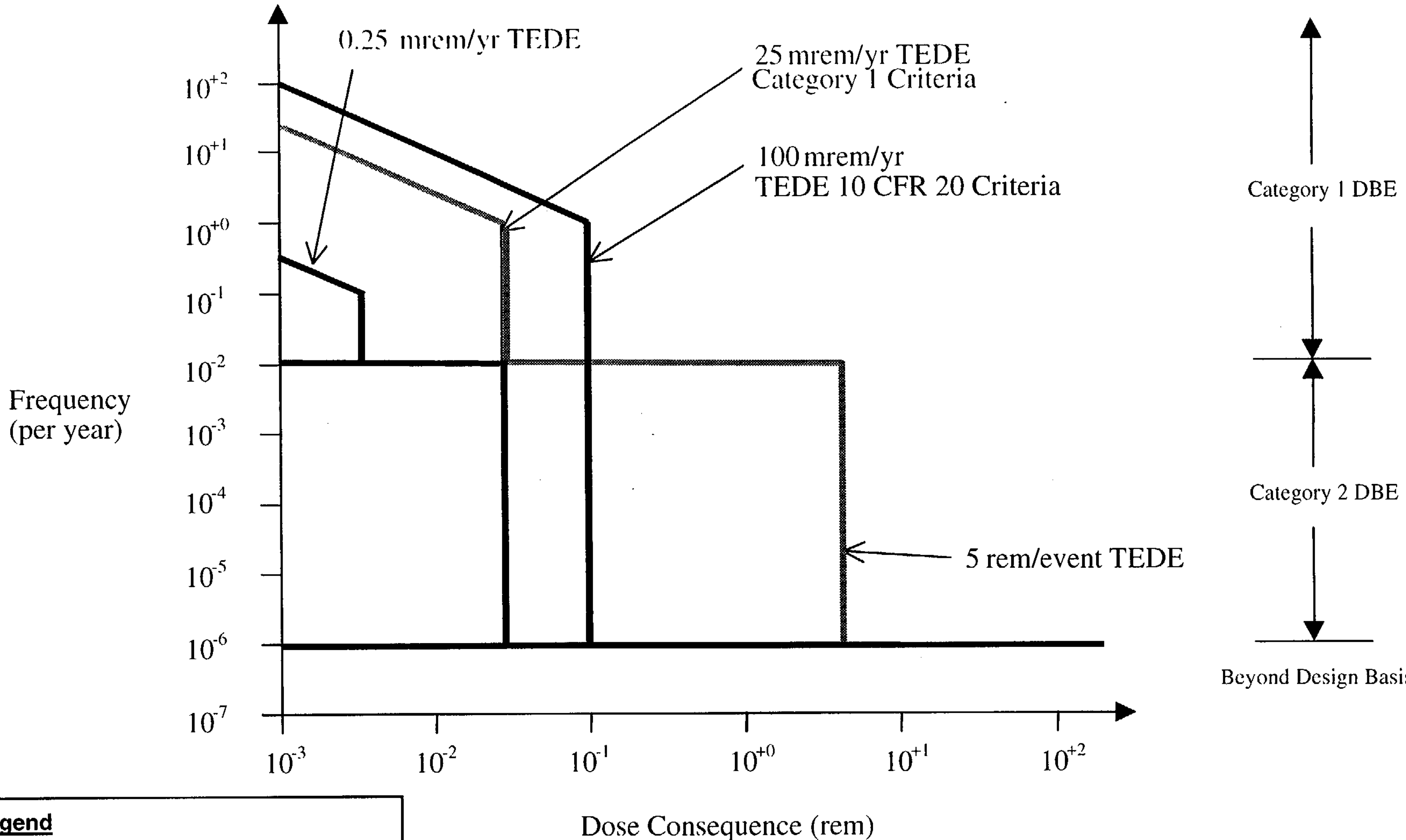
Information Derived from Experience Data

- Identification of potential hazards and estimates of event frequencies/probabilities
  - Generic probabilistic risk assessment databases (equipment, human actions, common-cause)
  - Licensee Event Reports for fuel handling facilities
  - Crane data: Newport News Shipping; NUREG-0612; vendor information
- NRC staff have accepted the use of mean frequencies for binning event sequences into Category 1, 2, or beyond design basis

Industry Precedents used for External Events

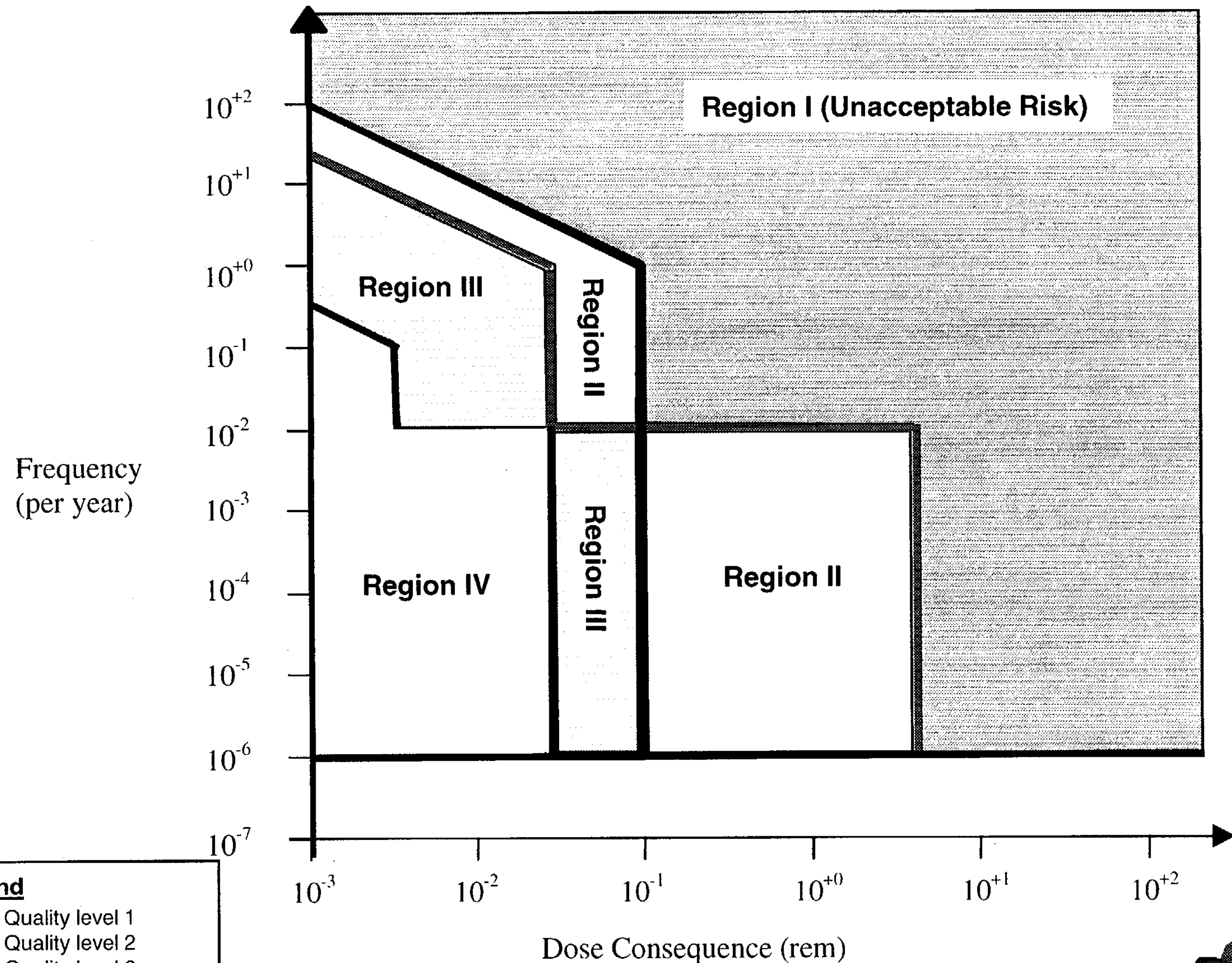
- Tornado/Wind
  - Tornado: NUREG 0800; ASCE 7.95
  - Wind: NUREG/CR-4461 (Tornado Climatology, 1986)
- Aircraft Hazards
  - Applicable portions of NUREG-0800 and DOE Standards (DOE-STD-3014-96)
    - Private fuel storage ongoing licensing activities
- Floods
  - ANSI/ANS-2.8-1992 - criteria for design basis flooding for nuclear safety-related features at power reactor sites

Proposed Criteria for Risk-Informed Classification of Structures, Systems, and Components



Legend
TEDE - Total effective dose equivalent
DBE - Design basis event
rem - measure of absorbed dose
mrem - millirem

Proposed Criteria for Risk-Informed Classification Analysis



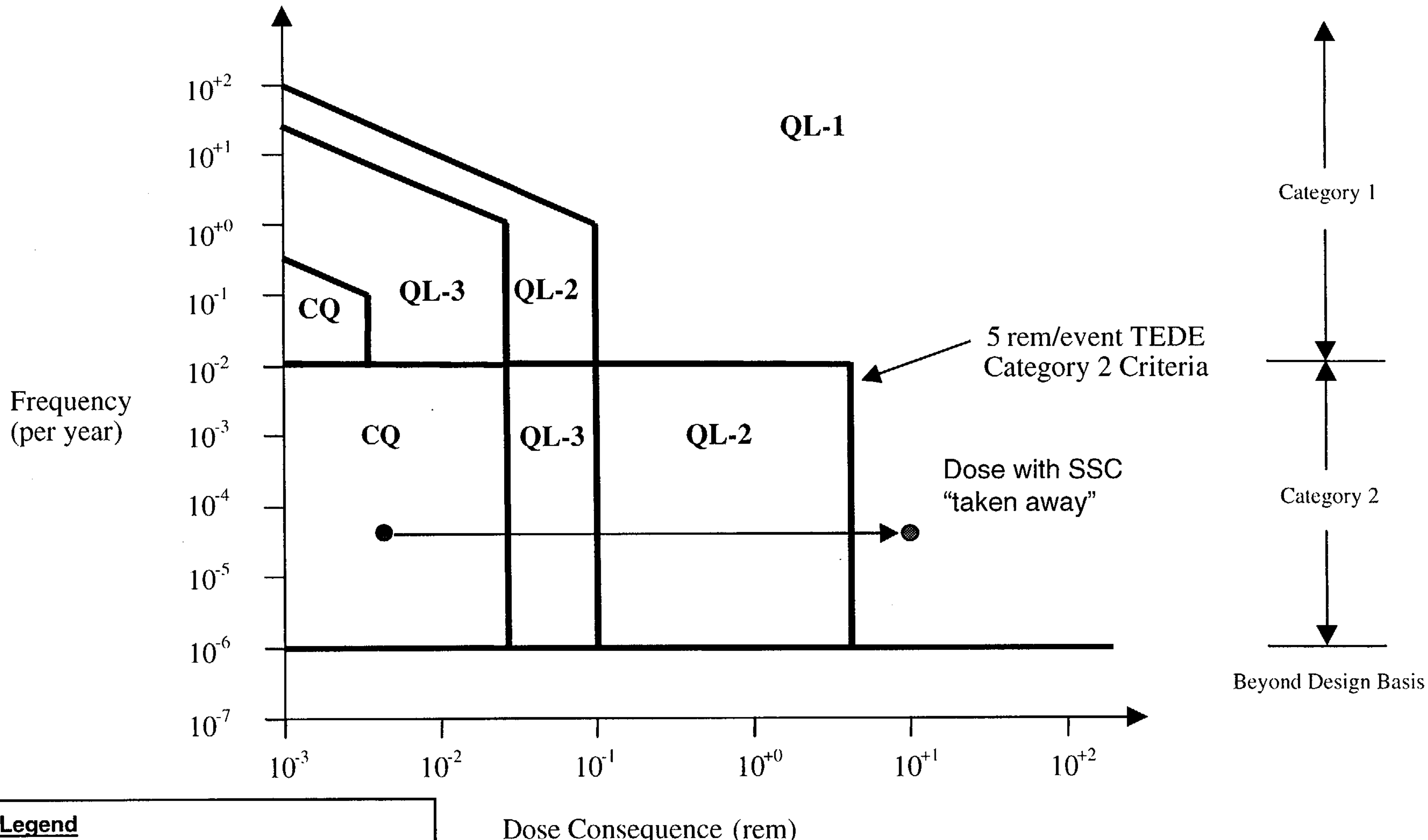
NOTE:
Region I = QL-1
Region II = QL-2
Region III = QL-3
Region IV = CQ

Legend
QL-1: Quality level 1
QL-2: Quality level 2
QL-3: Quality level 3
CQ: Conventional quality

Classification

- Structures, systems, and components selected as "Important to Safety" will be graded into 3 quality levels (QL)
  - QL-1: equivalent to reactor "nuclear safety related"
  - QL-2: graded QA program
  - QL-3: graded QA program
  - If not QL-1, 2, or 3 then commercial quality
- "Important to Safety" in proposed 10 CFR 63 includes components that would be in an augmented quality assurance program at reactor facility
  - A graded quality assurance program will be used to appropriately apply the 18 criteria from 10 CFR 50 Appendix B
  - Both deterministic and risk criteria will be used in classification of structures, systems, and components

Example of Structures, Systems, and Components Important to Safety



Legend
CQ - Commercial quality
QL - Quality levels
rem - Measure of absorbed dose
SSC - Structure, system, and component
TEDE - Total effective dose equivalent

Summary
Integrated Safety Analysis Process

- **Yucca Mountain's proposed integrated safety analysis is risk-informed**
  - Hazards assessment
  - Quantified frequency of event sequences
  - Event sequences categorized by frequency into Category 1, 2, or beyond design basis events
  - Uncertainty assessments or sensitivity studies performed
  - Structures, systems, and components classified based on risk insights
- **Integrated safety analysis will use nuclear industry precedence, experience, and data**
  - External event hazards draws from nuclear precedent
  - Historical information and data will be used to identify initiating events and support quantification of failure rates

Overview of Recent Engineering Developments

- **Describe the work to date to support the need for flexibility in the site recommendation design**
  - **Discuss Yucca Mountain's methodology for thermal management**

Site Recommendation Design

- Prior engineering work focused on hotter designs
- Yucca Mountain Science and Engineering Report identified a "Higher Temperature Operating Mode" and five example "Lower Temperature Operating Modes"
- Four of five low temperature modes keep average waste package maximum surface temperature below 85°C

Recent Engineering Developments
(Continued)

- Total System Performance Assessment-Site Recommendation was based upon higher temperature mode
- FY01 Supplemental Science and Performance Analysis included results based upon a low temperature mode-somewhat different from five example modes

Present Design Methodology

- Single design
- Range of operational variables
- Combining the single design with various combinations of operations variables results in specific thermal responses
- The range of thermal responses will be bounded by the Total System Performance Assessment and Integrated Safety Analysis for license application

Single Site Recommendation Design

- **The design presented in the Yucca Mountain Science and Engineering Report and the FY01 Supplemental Science and Performance Analyses document represent a single design and will work over the full range of thermal modes**

Design Parameter	Fixed for SR for all Modes Analyzed
Repository capacity	70,000 MTHM
Emplacement rate	3,000 MTHM/year after 5-year ramp-up
Emplacement period	~23 years
Sequence of waste package emplacement	interspersed hotter and cooler packages to achieve average linear power desity
Waste package design	Large, horizontally emplaced packages with corrosion-resistant outer shell of Alloy 22 and structural inner shell of 316NG
Number of waste packages	~11,000
Initial waste package power	11.8 kW maximum
Drift diameter	5.5 meters
Drift spacing	81 meters
Drip shields	Titanium, continuous

Potential Range of Operational Variables

These variables and ranges are preliminary and will continue to evolve as the Total System Performance Assessment is refined and the design is optimized.

Variable	Lower Limit	Upper Limit
Associated with the Waste Stream		
Assembly Age at Arrival	5 years	N/A
Assembly Burn-up	N/A	75 GWd/t
Assembly Enrichment	N/A	5.0%
Associated with Repository Operations		
Ventilation Duration	0 years	300 years
Ventilation Rate	0 CMS	15 CMS
Waste Package Average Spacing	0.1 meters	8 meters
PWR WP Capacity	1 assembly	21 assemblies
BWR WP Capacity	1 assembly	44 assemblies
Duration of Assembly Staging at Site	0 years	30 years
Amount of Staging at Site	0 MTHM	40,000 MTHM
Waste Package Power at Emplacement	N/A	11.8 kW
Areal Mass Loading	25 MTHM/acre	60 MTHM/acre
Linear Line Loading	N/A	1.45 kW/m

Options Considered for Site Recommendation

	S&ER[a] Design (TSPA Rev 0)	Scenario 1 Increased Waste package Spacing and Extended Ventilation	Scenario 2 De-Rated or Smaller Waste Packages	Scenario 3 Increased Spacing and Duration of Forced Ventilation	Scenario 4 Extended Surface Aging with Forced Ventilation	Scenario 5 Extended Natural Ventilation	SSPA[b] HTOM	SSPA[b] LTOM
Design Parameters								
Drift Center-to-Center Spacing (m)	81	81	81	120	81	81	81	81
Number of Waste Packages	~11,000	~11,000	~15,000	~11,000	~11,000	~11,000	~11,000	~11,000
Operational Parameters								
Average Waste Package Spacing	0.1	2	0.1	6	2	0.1	0.1	1.2
Surface Aging (years)	0	0	0	0	~~0~~ 30	~~20~~ 0	0	0
Emplacement Period (years)	~23	~23	~23	~23	~~~243~~ 50	~23	~23	~23
Forced Emplacement Ventilation After Start of Emplacement (years)	50	75	75	125	125	75	50	300
Natural Ventilation (years)	0	250	250	0	0	>300	0	0

[a] Source: Yucca Mountain Science and Engineering Report, DOE/RW-0539

[b] Source: FY01 Supplemental Science and Performance Analyses, Vol. 1 and 2

Legend
TSPA - Total System Performance Assessment
HTOM - High Temperature Operating Mode
LTOM - Low Temperature Operating Mode

Options Considered for Site Recommendation

(Continued)

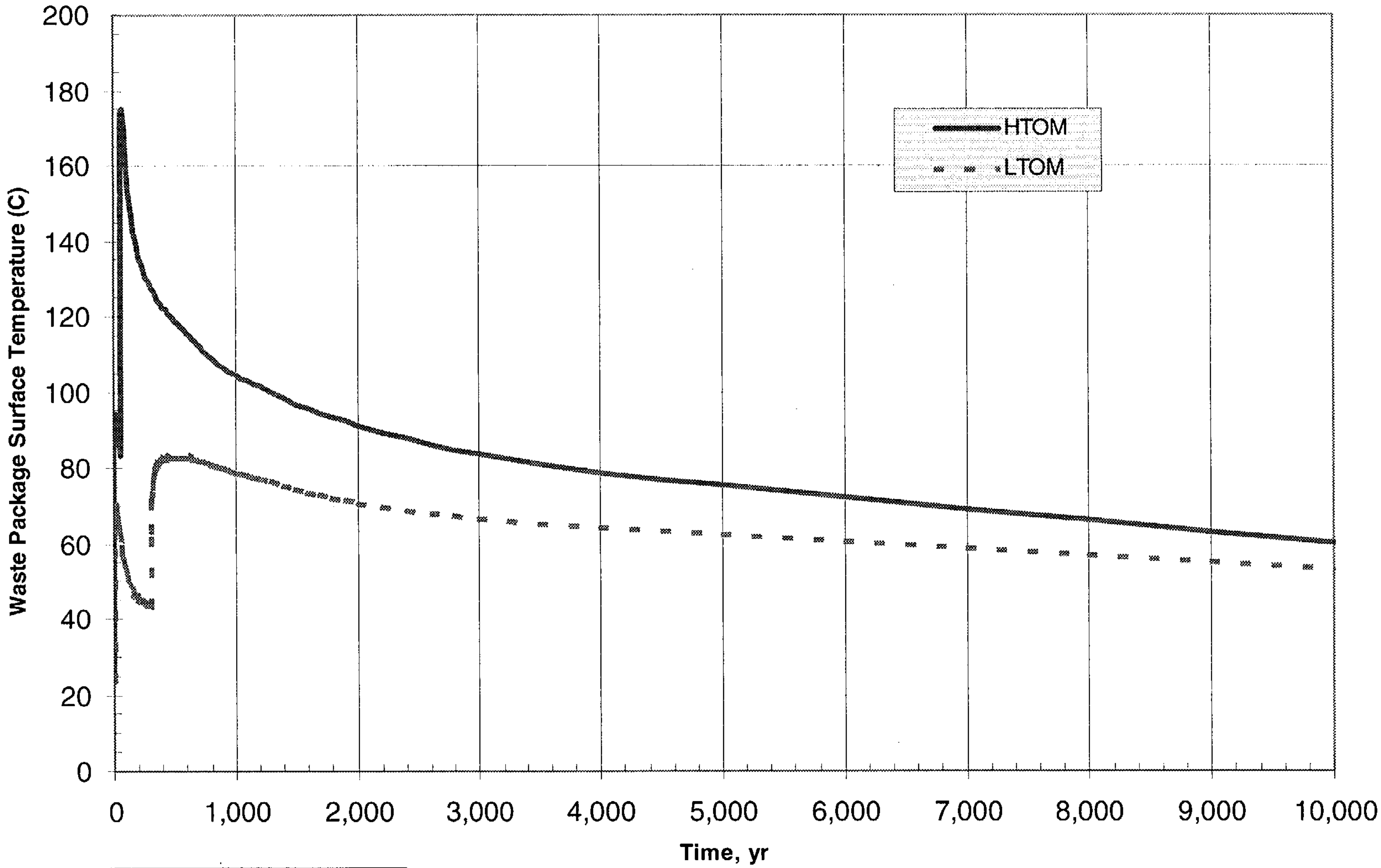
		Scenario 1	Scenario 2	Scenario 3	Scenario 4	Scenario 5		
	S&ER[a] Design (TSPA Rev 0)	Increased Waste package Spacing and Extended Ventilation	De-Rated or Smaller Waste Packages	Increased Spacing and Duration of Forced Ventilation	Extended Surface Aging with Forced Ventilation	Extended Natural Ventilation	SSPA[b] HTOM	SSPA[b] LTOM
Results								
Linear Thermal Loading (kW/m)	1.45	1.00	1.00	0.7	0.5	1.45	1.35	1.12
Total Emplacement Drift Excavation Length (km)	~60	~80	~90	~130	~80	~60	~60	~75
Required Emplacement Area (acres)	~1,150	~1,600	~1,800	~2,500	~1,600	~1,150	~1,150	~1,450
Average Waste Package Maximum Temp (°C)	~160	<85	<85	<85	<85	~~<85~~ <96	~160	<85

[a]Source: Yucca Mountain Science and Engineering Report, DOE/RW-0539

[b]Source: FY01 Supplemental Science and Performance Analyses, Vol. 1 and 2

Legend
TSPA - Total System Performance Assessment
HTOM - High Temperature Operating Mode
LTOM - Low Temperature Operating Mode

Thermal Response



Legend

HTOM - High Temperature Operating Mode
LTOM - Low Temperature Operating Mode

Attributes of Flexible Operations

- Operate under various thermal modes
- Accommodate variations in waste stream characteristics
- Accommodate changes in receipt and emplacement rates
- Provide for checks and balances for problem detection and corrective action

Licensing Approach with Respect to Flexible Operating Modes

- License application will be for a single design that incorporates the appropriate structures, systems, and components and operations philosophies that will allow safe acceptance, processing, and emplacement of waste
- Operational parameters will be bounded and included in the license specifications
- Thermal response will be bounded by the design parameters and operating modes analyzed in the Total System Performance Assessment and Integrated Safety Analysis for the license application

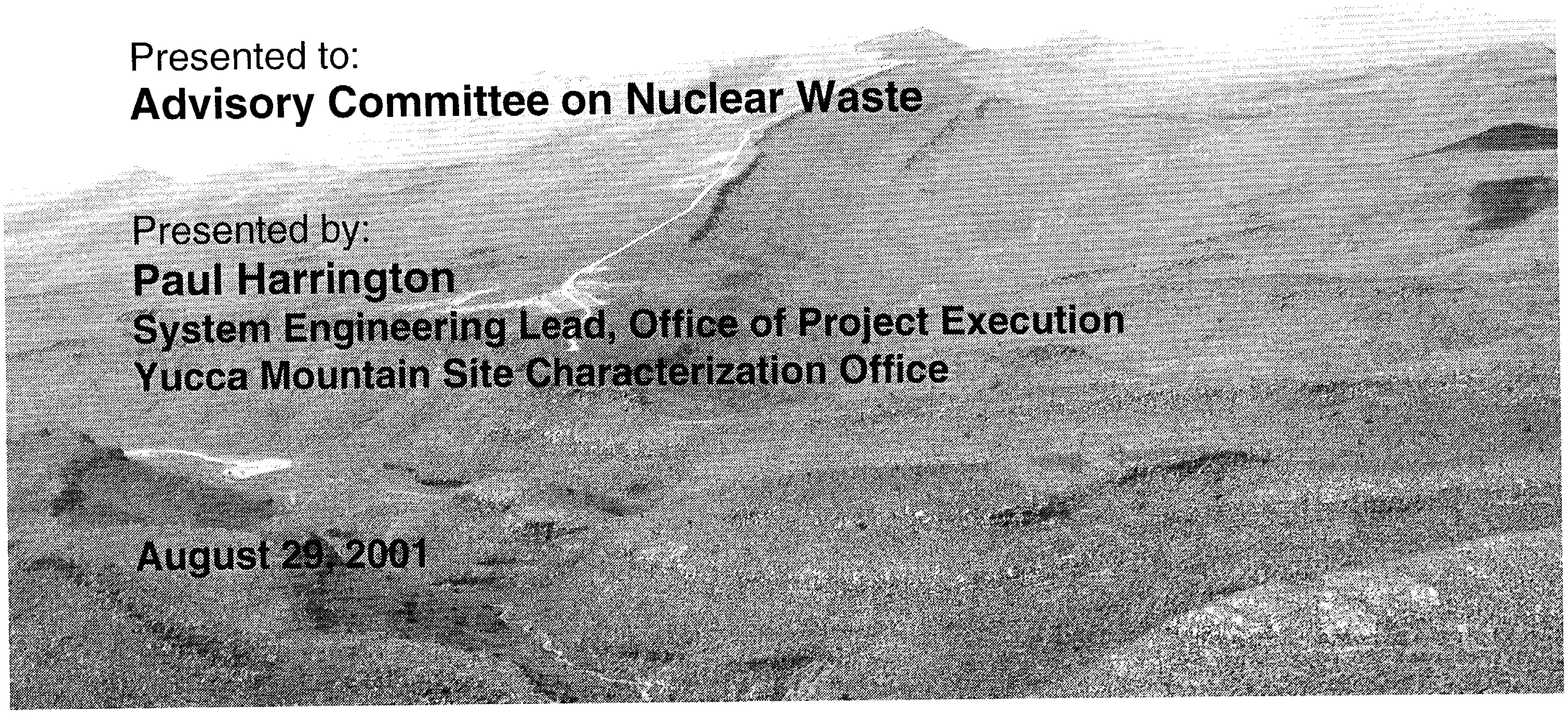
Summary
Recent Engineering Developments

- Design and operational parameters have been identified for site recommendation engineering and performance analyses
- The ongoing design and performance analyses provide a foundation that will allow us to continue to converge on optimum design and operational parameters during the licensing phase of the program
- License application will be for a single design, with operational parameters specified in the license specifications and bounded by the Total System Performance Assessment and Integrated Safety Analysis for the license application

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

YUCCA MOUNTAIN PROJECT

DOE-Yucca Mountain Preclosure Plans



Presented to:
Advisory Committee on Nuclear Waste

Presented by:
Paul Harrington
System Engineering Lead, Office of Project Execution
Yucca Mountain Site Characterization Office

August 29, 2001

Overview

- Introduction and objectives of the technical exchange
- Structure of NRC preclosure safety area
- Preparation for the technical exchange
- Items discussed at the technical exchange
- Meeting results
- NRC/DOE agreements
- Summary

Introduction and Objectives of the Technical Exchange

- Technical exchange was one in a series of prelicensing meetings related to NRC issue resolution and sufficiency review
- First technical exchange related to the preclosure safety area where NRC/DOE agreements were made
- Reach agreement on the structure of the preclosure safety area
- Discuss a limited number of issues under selected preclosure safety topics

Structure of NRC Preclosure Safety Area

- To track the status and agreements reached NRC has broken the preclosure safety area into ten topics identified in a Reamer letter dated April 27, 2001
- No preclosure safety topics Issue Resolution Status Report has been issued
- NRC will further define subtopics and/or acceptance criteria to be addressed in a license application
- Subtopics and/or acceptance criteria will be outlined in future preclosure meetings and the Yucca Mountain Review Plan
- The April 27, 2001, letter identified the subtopics to be discussed at the first technical exchange

Preclosure Safety Topics

- Site description
- Description of structures, systems, components, equipment, and operational process activities*
- Identification of hazards and initiating events*
- Identification of event sequences*
- Consequence analysis*
- Identification of structures, systems, and components important to safety; safety controls; and measures to ensure availability of the safety systems*
- Design of structures, systems, and components important to safety and safety controls*

*Indicates topics for which subtopics discussed at technical exchange

Preclosure Safety Topics
(Continued)

- Meeting the 10 CFR Part 20 as low as is reasonably achievable requirements for normal operations and Category 1 event sequences
- Plans for the retrieval and alternate storage of radioactive wastes
- Plans for permanent closure and decontamination, or decontamination and dismantlement of surface facilities

Preparation for the Technical Exchange

- NRC staff provided talking points, preliminary comments, preliminary acceptance criteria and preliminary draft staff positions on issues
- DOE prepared preliminary responses

Items Discussed at the Technical Exchange

- Integrated safety analysis process
- Aircraft crash hazard
- Tornado missiles
- Events screened out by design
- Justification of probability estimates
- Consequence analysis
- Identification of structures, systems, and components important to safety
- Identification of items important to waste isolation

Items Discussed at the Technical Exchange
(Continued)

- **Differentiated approach to providing information in the license application**
- **Preclosure criticality and burnup credit**
- **Waste package design and fabrication**

Meeting Results

- Following the discussions at the technical exchange, NRC stated that it agreed with DOE's general Integrated Safety Analysis methodology for
  - The identification of event sequences
  - Use of the mean value of probability distributions for event categorization
  - Consequence analysis
  - Identification and quality level categorization of systems, structures, and components important to safety

Meeting Results
(Continued)

- Following the DOE presentation the NRC stated that it was in general agreement with
  - DOE's proposed differentiated approach for information in the license application for a construction authorization and the license application to receive and possess nuclear waste
  - Level of design detail for systems, structures, and components to be included in the license application

Issues

- NRC stated that while the staff agreed with the DOE that utility reactor records are a more accurate and reliable indicator of commercial spent nuclear fuel burnup, the current NRC position was that physical measurement would be required to verify the reactor records for fuel assemblies for which burn-up credit was claimed. DOE believes that measurements of all such assemblies are not necessary
- DOE has completed a preliminary analysis of the aircraft crash hazard. NRC stated that detailed projections of future flight activity would be required in the license application. DOE believes that reasonable bounding assumptions should be acceptable. NRC agreed to review DOE's plan which will rely on available information for civilian and military aircraft

NRC/DOE Agreements

- **DOE to provide**
  - Plans for a comprehensive aircraft hazards analysis and to participate in a follow up meeting with the staff
  - A comprehensive tornado missile hazards analysis
  - Update of the procedure for identifying structures, systems, and components important to safety and important to waste isolation
  - Guidelines for conduct of the integrated safety analysis
  - Update to the Preclosure Criticality Analysis Process Report
  - An update of the Waste Package Design Methodology Report (two agreements)
  - An update of the Waste Package Operations Fabrication Process Report (two agreements)

NOTE: Total of nine agreements

Meeting Followup

- In addition to the formal NRC/DOE agreements
  - DOE will provide formal comments on the NRC draft staff position on risk significance categorization of structures, systems, and components important to safety
  - DOE will provide a white paper articulating the details of the Yucca Mountain quality classification and grading methodology

Summary

- DOE gained an increased understanding of NRC staff expectations of level of detail to be included in a license application
- DOE obtained valuable feedback on the acceptability of DOE's integrated safety analysis methodology and DOE's proposed differentiated approach for information to be included in the license application