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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON PLANT LICENSE RENEWAL

NIST NBSR REACTOR

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WEDNESDAY, FEBRUARY 4, 2009

ROCKVILLE, MD

The Subcommittee convened in Room T2B3 in the Headquarters of the Nuclear Regulatory Commission, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, at 8:30 a.m., John Sieber, Chair, presiding.

SUBCOMMITTEE MEMBERS PRESENT:

JOHN SIEBER, Chair

JOHN STETKAR

J. SAM ARMIJO

WILLIAM J. SHACK

SAID ABDEL-KHALIK

MICHAEL T. RYAN

OTTO L. MAYNARD

CHARLES H. BROWN, JR.

HAROLD B. RAY

DENNIS C. BLEY

NRC STAFF PRESENT:

PETER WEN, Designated Federal Official

TIM McGINTY

KATHRYN BROCK

WILLIAM KENNEDY

AL ADAMS

JOHNNY EADS

ALSO PRESENT:

ROBERT DIMEO

WADE RICHARDS

THOMAS MYERS

PAUL BRAND

MIKE ROWE

DAVID BROWN

ROBERT WILLIAMS

TABLE OF CONTENTS

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

(8:25 a.m.)

CHAIR SIEBER: The meeting will now come to order, and good morning, everyone. This is a meeting of the Plant License Renewal Subcommittee. My name is Jack Sieber. I'm Chairman of this subcommittee meeting, and today we will hear about the license renewal application of the National Institute of Standards and Technology reactor, which is located pretty close to here. It's at Exit 10, Montgomery Village exit, and you can see it from 270, so even though we've argued about how far that is, it's seven or eight miles from here, so it's convenient to us.

ACRS members in attendance are Otto Maynard, Bill Shack, Mario Bonaca, Said Abdel-Khalik, Sam Armijo, Charles Brown, Harold Ray, Mike Ryan, who isn't here quite yet, John Stetkar, and Dennis Bley. Peter Wen of the ACRS staff is the designated federal official for this meeting.

The purpose of the meeting is to review the license renewal application for the National Bureau of Standards Test Reactor, and, of course, National Bureau of Standards, the name has changed since this reactor was built, but it's still known as NBSR reactor, the Draft Safety Evaluation Report and the associated documents. We will hear presentations from representatives of the Office of Nuclear Reactor Regulation and the Applicant.

The subcommittee will gather information, analyze relevant issues and facts, and formulate proposed position and actions as appropriate for deliberation by the full committee. The rules for participation in today's meeting were announced as part of the notice of this meeting previously published in the Federal Register on April 15, 2008. We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

A transcript of the meeting is being kept and will be made available in the Federal Register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the subcommittee. Participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard.

As a introduction to this meeting, the ACRS has reviewed something like 52 applications for power reactor license renewal, and I've been here ten years, and this is the first in that period of time that we have had a research, test, or educational reactor for license renewal, even though, if you have, which I'm sure you have -- the introduction of the SER, the license for this reactor has been previously renewed, but this is one of the original reactors of this type.

This reactor operates at about 20 megawatts thermal, which is the highest power. I think there is only one other reactor that operates at that power, and that's at Brookhaven. On the other hand, the conditions in the reactor vessel and the adjacent systems are not what we would call harsh by power reactor standards. It operates at roughly 115 degrees Fahrenheit.

The pressure vessel is ASME code 150-pound pressure vessel, so compared to a power reactor there's a lot of unique features, and in addition to that, the licensure of this reactor falls under different parts of the Title X of the Code of Federal Regulations, and therefore the requirements are different, and so we need to keep in mind as subcommittee members the fact that the rules are slightly different. There is an SRP that the staff has developed that provides guidance to the staff as to how to do the review and to the Applicant as to what needs to be prepared, and so we should keep that in mind as we go through the presentations today.

I have, along with Peter, developed an agenda, which is more detailed in nature than we usually have for a meeting of this type, and that was intentionally done to provide the regulatory background and also highlight the areas that need our attention for license renewal of this reactor.

So, with that, I'd like to introduce Tim McGinty of NRR to introduce the staff, and I've asked the staff to make a short presentation on the regulations that apply to this reactor and the scope of their review. Tim?

MR. MCGINTY: Good morning, Mr. Chairman and Members of the subcommittee. My name is Tim McGinty. I'm the director of the Division of Policy and Rulemaking in the Office of NRR, and my division is responsible for the license renewal application before you today.

Sitting at the table with me is Kathryn Brock. She's Chief of the Research and Test Reactors Branch A. Also to my further left is Mr. William Kennedy. He's the NRC Project Manager for the renewal review, and he'll be leading the staff presentations this morning, and also behind me is Mr. Johnny Eads.

He's the Chief of the Research and Test Reactors Branch B.

In the audience we have various NRC staff who contributed to the review, including financial qualifications reviewers, the environmental review project manager, emergency planning reviewer, and technical review contractors from Washington Safety Management Solutions.

Today we will begin with a short presentation of the licensing history of the National Bureau and Standards Reactor, as the Chairman said, and the review criteria used in evaluating the licensee's renewal application. The licensee will follow this with their presentation, and after the break we will hear from Mr. Kennedy and Mr. Eads regarding the staff safety evaluation report and staff's inspection history.

I will point out that the staff safety evaluation does contain one open item, a discrepancy between the timing requirement in the regulations that in the licensee's requalification plan for administration of the operator requalification program. As Mr. Kennedy will explain, we expect to resolve this open item prior to the ACRS full committee meeting scheduled for April of this year.

With that, I'll turn the presentation over to Mr. Kennedy.

MR. KENNEDY: Thank you, Mr. McGinty, for your introduction. Good morning, Mr. Chairman, and members of the Advisory Committee on Reactor Safeguards Subcommittee. My name is William Kennedy. I am the project manager for the NIST license renewal before you today.

I'd like to thank the subcommittee for taking the time to scrutinize the staff's work, and my hope is that by the end of our meeting today, you will all have a clear understanding of what we did, why and how we did it, and what the bottom line is in terms of public safety.

Okay, so we're here today to talk about the National Institute of Standards and Technology National Bureau of Standards Reactor. As the Chairman mentioned, it's still called the National Bureau of Standards Reactor, despite the fact that NIST's designation has changed.

The topics I'm going to cover in the next 16 minutes are the licensing history of the reactor, the current licensing status, and the staff review criteria. Later this morning, after our break, we'll cover these topics except for staff inspection history, which will be covered by Mr. Johnny Eads.

So, licensing of this reactor began in 1961, when the National Bureau of Standards submitted a construction permit, application, and also an application for an operating license at 10 megawatts thermal power. This was for a heavy water, cooled and moderated reactor. In the next few bullets you'll see that I used terms like "believed" and "it was the opinion of," in referring to the ACRS, and that is the language that came right out of the letters that the ACRS provided to the Chairman of the AC at the time.

So the ACRS believed in 1963 that the proposed reactor, the National Bureau of Standards Reactor, could be constructed in Gaithersburg with reasonable assurance that it could be operated without any undue risk to the public health and safety, and following that, in 1963, the AEC did issue the construction permit.

In 1967, construction had been completed, and it was the opinion of the ACRS that the reactor could be operated as proposed without any undue risk to the health and safety of the public, and the Atomic Safety and Licensing Board concurred with the ACRS's recommendation and that of the regulatory staff at the time, and the AEC issued a provisional operating license, Number TR5, designated for Test Reactor 5.

The provisional operating license was, I believe, for a short term, 18 months or two years, and it was really in order to let the reactor get started up, achieve full power before taking a final look at everything and then issuing the full-term license, and the first criticality was achieved on December 7, 1967.

In 1969, the reactor reached full power and began operation at 10 megawatts on February 6, and in 1970, the ARCS reaffirmed its previous conclusion and again recommended conversion of the current provisional operating license to the full-term operating license, and that license was issued with a term of 15 years.

So, moving on to the previous license renewal in 1980, the National Bureau of Standards applied for a 20-year renewal, and they also applied for a power upgrade to 20 megawatts thermal. I believe the reactor had originally been designed for 20 megawatts, but it was not until this point that the upgrade was made. In 1984, the ACRS believed that there was reasonable assurance that renewal of the license may be granted without involving any undue risk to the health and safety of the public, and NRC did renew that license for a period of 20 years.

So, since the last license renewal, the NRC has issued three license amendments. One of these was in connection with the construction and installation of the NIST cold neutron source and the guide hall in order to ensure that in the case the reactor was operating that the guide tubes, which actually penetrate the building, the reactor building wall, could be isolated or would already be isolated.

Also, there was a change in the requirements for the primary heat exchangers, and this was done really to allow NIST more flexibility in replacement of their heat exchangers and upgrades. And at the -- it was at the same time was when the Department of Commerce decided to change the designation of the National Bureau of Standards to the National Institute of Standards and Technology. However, the reactor, again, did keep its original designation. There were also some administrative changes to the technical specifications.

So, currently, the licensee is operating the reactor under the provisions of 10 CFR 2.109, which deals with timely renewal. You're probably used to hearing a five-year time frame for timely renewal. We in the research and test reactors area have a 30-

day timely renewal, so NIST did file their application 30 days prior to when the license would have expired, which was May 16, 2004, and so they continue to operate under timely renewal under the normal inspection program.

About a year ago, we issued -- the NRC issued the Environment Impact Statement for license renewal, and there were, I believe, no -- all of the environmental impacts were deemed to be small in the same effect that any of the alternatives would have had, so there was really no different environmental impact from continuing operation of this reactor. And just his past month, we completed our draft Safety Evaluation Report of the renewal application, and as Mr. McGinty mentioned, we do have one open item, and I will come back to that later after the break.

Okay, so underpinning all of research and test reactor regulation is this idea of minimum regulation that's stipulated in the Atomic Energy Act, and this is done so that there can be a wide variety of research conducted at these facilities, and they will not be overburdened by regulators. However, we're still responsible for ensuring safety and security, and so we have to do our regulation consistent with our obligations but also keep it to the minimum necessary.

This comes out, and Part 50, the general design criteria, do not apply to research reactors. Also, Part 50, Appendix E on emergency planning, there's a statement in there that says the NRC will essentially apply the plan as necessary, planning standards as necessary. And as defined in Part 50, this reactor is a testing facility or a test reactor, and the criterion that makes it a test reactor in the case of NIST is that its power level is greater than 10 megawatts thermal.

And, so, Part 54, which you're all used to seeing, I'm sure, for power reactor license renewal, does not apply to non-power reactors, and this was a conscious staff decision when Part 54 was being crafted, and essentially there was already a license renewal process in place for research and test reactors at that time.

Also, research and test reactors are not required to submit updates of their FSAR, so we really go through an entire review of the facility when we do a license renewal. We don't -- we don't try to focus on just certain discrete areas. We look at everything again, much like we would in an initial application. So, our review, we primarily looked to see that the facility and the application satisfies the requirements of Part 20 for radiation protection and Part 50, especially in the area of the technical specifications.

Some of these facilities have not gone through license renewal for quite some time, and the regulations have actually changed, so part of the renewal process is to make sure that all of the technical specifications are up-to-date and in accordance with the regulations as they currently stand. Also, in respect to Part 51, environmental, I did mention that we have issued the Environmental Impact Statement.

Now, there are other parts of the CFR that do apply to research reactor license renewal or test reactor license renewal, but I've focused on these, because this is really where the majority of our work does -- where we do the majority of our work. There are other parts that apply to security, but I haven't -- I'm not going to discuss those today.

Also, the NIST reactor is unique, because it is the only one of the reactors that research reactor branch regulates that is subject to the requirements of Part 100. That's reactor site criteria, and the guidelines in Part 100 are for --

that are applicable in this case are for accident doses and establishing the sizes of the power reactor in terms of low population zone, and in test reactors we have a site boundary, and we have a controlled area, and we have a restricted area, which have different definitions, and the controlled area is really what lies inside the fence boundary around NIST, and their entire emergency planning zone does fall inside their fenced boundary, and so we would say that they do have control over that area because of all of the entrances to the campus are monitored by security, and so the licensee has the ability to control all of the activities that are happening within that area.

And I'll go over it a little more later, but in the case of accidents, NIST -- the accident doses actually are below the limits for members of the general public for normal operations, so they certainly do satisfy these guidelines in Part 100, as well.

CHAIR SIEBER: On the other hand, there's a lot of employees at the -- on the NIST campus, right?

MR. KENNEDY: Correct.

CHAIR SIEBER: In the thousands, right?

MR. KENNEDY: Okay, so our review guidance is NUREG 1537. The staff looks at Part 2, which is the guidelines for preparing and reviewing applications for all licensing of non-power reactors, non-power reactors being both research reactors and test reactors.

There is a Part 1, which is guidance for licensees or applicants on how to prepare their document and what type of information the NRC would like to see. This was put out in February 1996, and it provides us with review criteria, and it's really -- it has been our Standard Review Plan since its issuance.

It was designed to apply to all non-power reactors, both those that hadn't been licensed yet, those that needed license renewal, those that were going to need decommissioning, and it also deals with, I believe, high enrichment to low enrichment conversions.

So all of the review criteria in this document are really not applicable to all cases the NRC reviews, and also this Standard Review Plan does reference a lot of other documents that we pull in to conduct our review. These include other NUREGs, regulatory guides.

We have Division 2 regulatory guides and also do some limited application of some of the power reactor reg guides, also the American National Standards Institute/American Nuclear Society 15 series standards, which cover everything from technical specifications to some fuel utilization issues, emergency planning, environmental citing.

So, when I set out to do this review, I felt there were a couple specific criteria that were really of the greatest safety significance, so I've chosen to highlight those here. The first is that when we look into the technical specifications, we've always got to have a safety limit, and in the case of NIST, that's 450 degrees C for aluminum-clad fuel and want to make sure that in no case are we ever going to approach that temperature, both during normal operations and also during any credible accident scenarios.

The other is we have responsibility to protect the public from the effects of ionizing radiation, so we have to make sure that the applicant or the licensee in this case has satisfied all of the requirements of Part 20, and not only satisfied them, but also, you know, uses an ALARA program that makes sure that they're going to keep their doses as low as they can.

CHAIR SIEBER: I think it's important to recognize in the first sub-bullet there where it talks about credible accidents. There was an accident analyzed where there was total blockage to a fuel element which had power, which resulted in melting of that fuel element, and the purpose of postulating that was to calculate what dose would be created by that particular accident, but that is not a credible action.

There is no mechanism where that would occur in this reactor, and it's just done to estimate what the dose would be should some unforeseen circumstance that nobody has ever dreamed of occur that would cause fuel damage, and that is the worst fuel damage that you can have in this reactor, so there is that important distinction. That accident is discussed, but it is not credible.

MR. KENNEDY: Thank you for mentioning that. That ties in also to this third bullet, which is that in the case of this maximum hypothetical accident that the Chairman mentioned, we still see radiation doses that are a very, very small fraction of the guidelines in Part 100, and, as I mentioned, also well below the limits in Part 20 for normal operation.

MEMBER RYAN: Just for everybody's benefit, could you put a number to that, to those statements, how many millirem per year per accident? If you're going to get to it later, that's fine.

MR. KENNEDY: I will get -- I will get to it later, but I will tell you that at the 400 meter boundary, the whole body total effective dose equivalent is 7 millirem.

MEMBER RYAN: Thanks.

MR. KENNEDY: The Part -- the Part 20 limit for normal operation is 100 millirem per year.

CHAIR SIEBER: For the general public.

MR. KENNEDY: For the general public, yes.

CHAIR SIEBER: Right.

MEMBER MAYNARD: One thing I question a little bit on the specific criteria, and we may get into this later, this seems to be for an initial license would be significant. We're really looking at renewing a license. I don't see anything really about aging or continued operation with older equipment and stuff. Are we going to get into that a little bit later, or is the criteria strictly assuming everything is brand new and we're reviewing it?

MR. KENNEDY: We will get into that later. We do not assume that everything is brand new and we're getting into it, so we will cover some issues that we call prior use of reactor components, so I'll cover your concerns.

MEMBER MAYNARD: Because I would think that would be -- you know, for a specific criteria for a license renewal, I would think there would need to be something about --

CHAIR SIEBER: I did put in the agenda opportunities for discussion of that, because I share your concern, and I'm sure NISTA has addressed it, and so has the staff, but it is not a power reactor, so they don't --

MEMBER MAYNARD: And I understand that.

CHAIR SIEBER: Okay. So we'll get to it, I think.

MR. KENNEDY: I think I can show later that your concerns about aging do really ultimately get wrapped into one of these three bullets.

MEMBER RAY: Back to the accident that Jack mentioned, the hypothetical, the dose then assumes no other failure.

MR. KENNEDY: That is correct.

MEMBER RAY: It means your safety features, there's no bypass of confinement or whatnot.

MR. KENNEDY: That's correct.

MEMBER SHACK: Has anybody ever done a PRA for this thing?

MR. KENNEDY: Not to my knowledge.

MEMBER RAY: At some point, just to tell you where I'm going, the language in the SER mentioned redundancy and diversity, but at other times it sounded like they were single train. For example, on the detectors, radiation detectors, it describes it as if there is only one detector in each of the three ranges, and I just wasn't clear at all about what the assumptions were on single failure, for example.

I understand all the business about this isn't a power reactor and the requirements don't apply, but nevertheless I'm still interested in what is assumed about failures when engineering safety features are being called upon.

MR. KENNEDY: We assume single failure, and we use a deterministic approach in our review.

MEMBER RAY: So, for example, on confinement isolation, you would assume a radiation detector failure that actuates the isolation?

MR. KENNEDY: We would not assume the failure of a detector in the case of a fuel failure, as well. We would not look at those together.

MEMBER RAY: Okay, well, that's what I

meant when I asked the question about if you postulate the accident and then you say what the dose is, the accident is the only failure.

MR. KENNEDY: Correct.

MEMBER RAY: So it's what we call --

MEMBER BLEY: Might call an initiated. There is the one thing that happens, and that is your failure. You don't do that plus one other.

MEMBER RAY: Plus a detector failure, for example.

MR. KENNEDY: That's correct, and that's what's specified in our Standard Review Plan.

MEMBER RAY: Yes, I'm not questioning the criteria. I'm just trying to understand the setup here.

MEMBER BROWN: I seem to remember on the -- Mike, on your comment, that that was the -- the dose you talked about was to the public outside the boundary, and I seem to remember when I was looking at, what is it, the SER or something, that the dose to the operators or the personnel was something in the order of 4 rem or something like that for this. Is that the number I remember?

CHAIR SIEBER: Maximum dose.

MEMBER BROWN: Maximum dose.

CHAIR SIEBER: Right.

MEMBER BROWN: I was just trying to get a calibration out on both things. Who are the people that are there operating, what they would get, and then what you've got on the site boundary, as well.

MEMBER RAY: While we're on the point, the Chairman asked before how are members of the staff at NIST treated? Are they members of the public outside of the restricted area, treating them as workers or not with regard to the accident analysis?

MR. KENNEDY: They are outside the restricted area.

MEMBER RAY: They're members of the public?

MR. KENNEDY: Yes, they would be within the controlled area, so they are able to get those people that are within the campus out of the area.

MEMBER RAY: That's why I asked. I just wanted to be clear that they weren't being treated under worker dose limits, but they're under 100 millirem per year.

MR. KENNEDY: That's correct, yes.

MEMBER RAY: Okay.

MR. KENNEDY: And we also did look at kind of the surrounding of what buildings are near there. You know, if there's high-elevated buildings, it would be more susceptible to any kind of a plume.

CHAIR SIEBER: Just keep in mind at the peak there was 900 and some badged employees --

MEMBER RAY: Right.

CHAIR SIEBER: -- not all of which were operators, but those people would be treated insofar as the regulations are concerned as radiation workers.

MEMBER RAY: Sure, because they have access to the restricted area, because they've --

CHAIR SIEBER: They've been trained and --

MEMBER RAY: Right. Okay. Thanks.

MR. KENNEDY: If there are no more questions, that's it for my presentation now.

CHAIR SIEBER: Thank you very much. I think we can now move on to the presentation by NIST. While the folks are getting ready here, the very first item on the second page of the agenda is a -- is my way of talking about aging mechanisms in plant systems, and I'm sure both the applicant and the staff will address that.

MR. RICHARDS: Good morning, Mr. Chairman, members of the ACRS. My name is Wade Richards. I am the Group Leader for Reactor Operations and Engineering. Dr. Dimeo is the NCNR Director, and Dr. Rowe is the Special Advisor to the Director and also past Director of the NCNRS.

I'd like to introduce portions of the staff we will be using tonight, today. Dr. Williams, head of our Nuclear Analytical Section. Paul Brand. Dr. Brand is the Chief of Reactor Engineering. Mr. Myers is the Chief of Reactor Operations, and Mr. Brown is our Chief of Health Physics.

The agenda we're going to be using is the agenda that we were sent by the Committee. We'll be using this outline for the very limited time we have, so if there is anything on this outline that you folks want to eliminate or expand upon, we're ready to do that, and we'll -- Dr. Dimeo will present the background for the facility.

MR. DIMEO: Mr. Chairman, members of the ACRS, ladies and gentlemen, good morning. My name is Rob Dimeo, and I am the Acting Director of the NIST Center for Neutron Research. We are pleased to be here for this meeting on the relicensing of the NIST reactor.

The NIST reactor is the source of the NIST Center for Neutron Research, a national neutron scattering user facility serving over 2,200 researchers annually. The term "user facility" has a special meaning to us that's reflected in our mission, which is to assure the availability of neutron measurement capabilities to meet the needs of U.S. researchers from industry, university, and other government agencies.

Beam time for experiments is made available to the scientific community based on technical merit through an independent peer-reviewed proposal process. The research done at the NCNR is highly multidisplinarian, spanning basic and applied materials research to investigations into some of the most fundamental questions in nature.

In most of the research at the NCNR, neutrons are used to probe matter, and in some cases the neutron itself is studied as a laboratory. With greater than 300 publications, many of which appear in the highest impact technical journals, our scientific productivity is widely regarded as the highest of any neutron facility in the United States.

We are one of four major neutron scattering facilities in the U.S., the only one not run by the Department of Energy. Recent assessment of the neutron facilities found that, at least in terms of measurement capacity, the U.S. lags Western Europe by a significant margin. Reports from the White House and the American Physical Society emphasize a number of important observations regarding this essential measurement technique and the facilities that provide it.

First, NIST is the only facility providing a broad range, world class measurement capability. Second, NIST has the largest user program, mainly because of our unique cold neutron source, and third, the way to reduce the gap between the U.S. and Europe is to fully exploit the best neutron sources and increase the number beam lines and instruments.

A primary factor in the first two points is our excellent reactor and operations engineering staff, who continue to maintain and improve the plant in order to sustain the reactor's outstanding record of reliability. The last item shown points to the continuing national need for this measurement need.

Given the success of the NCNR --

MEMBER ABDEL-KHALIK: I have a question.

MR. DIMEO: Yes, sir?

MEMBER ABDEL-KHALIK: How proposed experiments are actually reviewed before they are approved.

MR. DIMEO: Sure. There's effectively two types of reviews that take place. The first is for a safety review for the samples that are going to be coming to the facility, and so that's in conjunction with our safety representatives in health physics.

The second review is an independent panel review given by our so-called Beam Time Allocation Committee. It's an independent panel of experts from various areas of science who essentially -- we have a call for proposal twice a year, so we'll get about 600 per year, and these will go through our Beam Time Allocation Committee for technical feasibility and, most importantly, technical merit.

MEMBER ABDEL-KHALIK: Are there any experiments that could potentially impact the core reactivity, and how are those reviewed?

MR. DIMEO: We have a -- I'm going to refer this to Wade, but the answer is no, and all experiments are reviewed by our health physics and reactor operations group, as well.

MR. RICHARDS: The experiments that are performed at the NVSR are mainly experiments in the beam tubes. We have very few experiments that go inside the core area.

MEMBER ABDEL-KHALIK: You have no control over what people would propose.

MR. RICHARDS: The experiments themselves in the beam tubes are actually reviewed before they're performed by the Safety Evaluation Committee, by the Hazards Review Committee, and then they are approved by the Director. No experiment is ever done out of the beam tube unless it has gone through the proper reviews, and it's reviewed for industrial safety, radiation safety, and nuclear safety.

CHAIR SIEBER: Strangely enough, I found a fairly good explanation of the types of experiments that can be done and are done in Mark's Mechanical Engineering Handbook, strangely enough, in my desperate search to find out what you do, so I'll give you that as a reference.

MR. DIMEO: So, given the success of the NCNR, its international reputation for excellence, and its critical role in the NIST mission, it's not surprising that it has received and continues to receive very strong support from senior leadership at NIST and the Department of Commerce to operate the NIST reactor cost-effectively while assuring the safety of the staff and the general public.

I could give you numerous examples of the support, but I've just listed a few here on the slide, and an example of the agency's strong commitment to the NIST Center for Neutron Research is that we are in the midst of an initiative to significantly expand our cold neutron measurement capability over five years as part of the America Competes Act. NIST and the Department of Commerce will continue to remain committed to the safe, reliable operation of the NIST reactor.

MR. RICHARDS: I will start out on the agenda that you put together. There is no way that I can be an expert on all these things, so I will be handing off to various members of the staff if I get stuck, so please bear with me here.

As you heard, the National Bureau of Standard's Reactor is a heavy water D_2O-moderated and cooled reactor. It's enriched fuel. It's a tank-type reactor designed to operate at 20 megawatts. It's a custom-designed variation of the old Argonne CP5 class of reactor.

The operating history is shown here. In 1984, as Bill mentioned, we increased the power to 20 megawatts. Since 1984, the power increased. The shims have been replaced three times, approximately every four years. We have cadmium-type shims that rotate, a semi-4 type. At 20 megawatts, they do burn out, so every four years we have to replace them, so they have been replaced at least three times since the last renewal.

In 1984, `94 -- I'm sorry -- the guide hall and cold source were installed and constructed, and in 2004 the application for license renewal was submitted. To date, the reactor runs on a 24/7 schedule on a 38-day cycle with normally ten days down for maintenance, and we annually -- the annual medium full-power days are about 147 -- 247. Excuse me.

CHAIR SIEBER: How long will a given fuel assembly stay in for? The only -- your fuelings, even though they're frequent, change just a small fraction of the core.

MR. RICHARDS: We change out four of our elements every fuel cycle, and every element goes through anywhere from seven to eight cycles, and each cycle is 38 days long.

CHAIR SIEBER: So that would be 16 months, 18 months lifetime for a fuel assembly.

MR. RICHARDS: Right.

CHAIR SIEBER: Okay. Now, you have here that you've changed heavy water three times, and I understand that was done to reduce radiation dose to workers.

MR. RICHARDS: We have a tech spec limit on the curies per liter in our heavy water, and when it starts approaching -- well, long before it starts approaching that limit, we will change out the heavy water.

CHAIR SIEBER: And even though I talked before the meeting, can you tell me about what the dose reduction has been?

MR. RICHARDS: For the heavy water changeout?

CHAIR SIEBER: Yes.

MR. RICHARDS: Dave, could you give us some idea?

MR. BROWN: What was the question?

CHAIR SIEBER: What dose reduction do operators currently achieve by every changing out of heavy water?

MR. BROWN: By changing out the heavy water? My name is Dave Brown, health physics in VSR. I would say at the peak, operators receive 200 to 300 millirem per year from tritium exposure, and when we change it out, we start with a fresh batch of D_20 that has a very low tritium concentration and then typically would reduce that level to maybe 50 millirem a year total.

CHAIR SIEBER: So that's pretty effective.

MR. BROWN: Pretty effective.

CHAIR SIEBER: Pretty expensive.

MR. BROWN: Very expensive.

CHAIR SIEBER: Thank you.

MEMBER ABDEL-KHALIK: How do you dispose of the tritiated heavy water?

MR. RICHARDS: That water is sent up to ACL in Canada. They use it in their power reactors. It's rather clean water for them, 5 curies per liter. It doesn't bother them at all, but it's not that clean for us.

MEMBER ARMIJO: Where do you -- what happens with your spent fuel? Do you have a equivalent of a fuel pool in a power plant that you store it for a while and then ship it off somewhere? Exactly what happens?

MR. RICHARDS: We have a refueling system every cycle when we pull four elements. We pull four elements, put four elements in. The four elements that we pull out are transferred down to our storage pool, and about every five years, five to seven years, we will actually do a shipment to Savannah River.

MEMBER ARMIJO: Okay, so over a period of five years, how much fuel do you have in that storage pool?

MR. RICHARDS: How much fuel?

CHAIR SIEBER: Four elements --

MR. RICHARDS: Four elements.

CHAIR SIEBER: -- every 39 days or 59 days.

MEMBER ARMIJO: A lot of elements, more than what's in the core.

MR. RICHARDS: Oh, yes. Yes. There's 30 in the core. Do you know how many you have in your --

MR. MYERS: Yes, I'm Tom Myers, Chief Reactor Operations. We generate 28 elements a year, so four years you're looking just under 100 elements or so. We ship off typically 126 assemblies every five years.

MEMBER ARMIJO: Okay.

MR. RICHARDS: Following the outline, the primary cooling system is designed to transfer the 20 megawatts thermal heat from the core to the secondary system, of course. The nominal operating values for our primary system, 9,000 gpm. Now, these numbers are going to sound absurd to you people in the power reactor world, but 9,000 gpm, about 100 degrees Fahrenheit inlet, 114 degrees outlet.

The primary system is not pressurized. The discharge pump is about 65 psig. I think someone was saying something about the pressure vessel. We don't have a pressure vessel, actually, and the pressure operating psi is 7 psi?

MR. ROWE: Approximately. It's just a static head, micro. It's simply the static head of the D_2O, which is of the order of 7 psig.

CHAIR SIEBER: The pressure vessel, seven pounds is perfect.

MR. RICHARDS: The primary cooling system consists --

MEMBER BROWN: Excuse me. What's the ultimate heat sink for this?

MR. RICHARDS: The heat sink?

MEMBER BROWN: Yes, what's the ultimate heat sink?

MR. RICHARDS: We have a cooling tower.

MEMBER BROWN: Cooling towers.

MR. RICHARDS: Yes. The primary cooling system consists of pumps, heat exchangers, piping and valves, as it normally would. All components in our primary system are either aluminum or stainless steel. All of the materials in the system were certified and inspected in accordance with the federal specifications at the time, industrial standards, codes that were in existence at that time. All ASME codes, pressure vessel codes, ACI codes and all these were all part of the initial construction specifications.

MEMBER SHACK: I just -- when Brookhaven did a review of a reactor that was built roughly sort of same time, they -- for a seismic analysis they found that -- you know, they did some component. They had some tanks that weren't really seismically qualified. Their control room enclosure was an unreinforced cinder block structure, and they had some non-critical components that could fail and fall on critical components. Have you ever done a seismic walk-down looking for that kind of stuff in your reactor?

MR. RICHARDS: We did a seismic study on an experimental enclosure not too long ago, and I'm not aware of -- maybe Dr. Bley --

MEMBER SHACK: No cinder block structures anywhere inside the confinement building?

MR. RICHARDS: Confinement building is not cinder block.

MEMBER SHACK: No, not the cinder -- inside, partition walls of some sort.

MR. RICHARDS: I don't know the answer to that question.

MEMBER SHACK: Unreinforced walls of any kind inside?

MR. ROWE: There are some unreinforced

walls. I'm trying to remember exactly where. I think the answer to your question is formally we have not done the walk-down that you're asking about.

MEMBER STETKAR: I thought I read somewhere in there that the nominal -- you don't call it a safe shutdown earthquake, but the nominal seismic capacity of the facility is something around a tenth of a g. is that correct?

MR. RICHARDS: That's correct.

MEMBER STETKAR: Is that -- have actual formal qualifications been done for the electrical and mechanical structural equipment to a tenth of a g loading?

MR. RICHARDS: I'm not aware of any studies.

MR. ROWE: They were done in the beginning. At the initial licensing hearing that question was raised, and it was at that time that the architect, engineers, and designers of the facility stated that they -- not believed but that the building was satisfactory to .1g.

MEMBER STETKAR: That's a confinement building, for example.

MR. ROWE: Correct.

MEMBER STETKAR: Okay. Thank you.

MR. RICHARDS: This slide shows the 30 -- I don't have pointer, but this slide shows the 30 fuel elements, the shim arms, and the large cold source. It also shows the central plenum that comes up through the center of the core. The water comes up through the bottom of the elements and then comes down on the sides of the two outer plenums. I'm sorry. I went the wrong way.

CHAIR SIEBER: Have you ever had a shim rod sticking or sluggish in operations?

MR. RICHARDS: No. Our shim rods have been -- shim arms have been very reliable. I think we had one that got some water in it and swelled, but it didn't stick or anything. We just took it out, but that was many moons ago.

CHAIR SIEBER: But the failure of any one does not affect the --

MR. RICHARDS: No.

CHAIR SIEBER: -- your ability to shut down according to the assumptions in your safety analysis.

MR. RICHARDS: That's correct. Fuel element design and construction. The MTR type fuel elements have got a 50-year history of reliable use in many facilities, and that's what we have, an MTR type element. I don't know if I could bring up the back-up material or not, but there's a picture of the fuel element.

It has a -- it's a split-core design. It has 17 plates in the upper section and 17 plates in the lower section, and these two sections are separated by a seven-inch gap, which provides the thermal flux trap that our beam tubes are all on the center line, so these fuel elements have fuel in the top, the bottom, the center. The beam tubes look right at that center so that we can maximize the thermal flux, because this is a beam tube reactor facility.

We did change the composition of the fuel. We went to U_3O_8 and aluminum dispersion in 1991, and we also went from a 300-gram element to a 350-gram element of U_{235}. We are enriched to 93 percent.

Again, they have the 30 elements on a 38-day cycle. Four are removed every fuel cycle. The other 26 are relocated, and then the -- and the average burnup per cycle is around 69 percent, and roughly 7.4 kilograms of U_{235} at the start of core life, and at the end of core life we have about 6.4 kilograms.

CHAIR SIEBER: Just a quick question there. You know, having not seen how you do that, it sounds like an awful lot of fuel movement. Have you analyzed, you know, any kind of accident scenarios? You take four fuel elements out, and if I read the documents right, you relocate quite a large number of other fuel elements.

MR. ROWE: Yes. In the accident analysis we looked at the question of whether a fuel element was put into an incorrect location, and what was done was to take a fresh fuel element and put it in every possible location in the core and verify that everything was all right, that we remained within the critical heat flux ratio agreement.

In addition, where this reactor is refueled and operated, we don't have open places. We have one element moved at a time, so there is no way to sort of drop a fuel element into an open position. We have one open position as we're refueling, and that's the one we use. So the answer is yes, that has been analyzed.

MEMBER RYAN: That sure helps with -- you can't -- it's hard to put one in the wrong place, because there's only one place to go, but how about handling mishaps with, you know, I don't want to say dropping them but, you know, having them not handled correctly or having difficulty removing and replacing fuel elements?

MR. ROWE: I'm going to ask Tom to help my memory on this one. Dropped fuel element, have we had one, as I recall, Tom, Tom Myers?

MR. MYERS: Yes, since we rebuilt the head in `94, we've not dropped any fuel elements. Prior to that, possibly every two or three years an element would be dropped. When I say dropped, the element comes off the tool and drops maybe six feet at most.

MEMBER RYAN: Yes, that's the kind of thing I was thinking about.

MR. MYERS: It's relatively robust, but it's also important to remember that we do not refuel until at least five days have lapsed, so there is no chance that the element is going to be without cooling, that the padding is going to fail. Also, that distance that it's dropped is not going to be sufficient to cause any kind of a cladding breach.

MEMBER RYAN: Okay. Thanks.

MR. MYERS: And then we have ways of retrieving that element.

MEMBER RYAN: Tell us a little bit more about that after `94 you have not done that. That has not happened?

MR. MYERS: We have not. We rebuilt the --

in the center top plug of the core, there's a refueling plug in which many of the tools for moving the elements around are located. We don't pull the plug off and look down into the vessel and refuel. We use a mechanical maze. We use that mechanical maze to position the tools and the elements together.

MEMBER RYAN: Right.

MR. MYERS: When we pulled that head out in `94-`95, we rebuilt all the tools, and since then we have not dropped any fuel elements.

MEMBER RYAN: Okay.

MR. MYERS: It was more of an issue of the tools had become, you know, 30, 40 years, and it was time to rebuild them. Since then, there has been no problem.

MEMBER RYAN: Great. Thank you very much.

CHAIR SIEBER: Just as a matter of clarification, you talked about a seven-inch gap in the fuel assembly. That's a gap in special nuclear material. The structure of the fuel assembly is continuous throughout its length. Is that not true?

MR. ROWE: The outer part of the fuel element is continuous, but the plate's actually solid.

MEMBER ARMIJO: So that you have something like a duct or something holding the upper part of the fuel element to the lower part?

MR. ROWE: Yes.

MEMBER ARMIJO: It's one.

MR. ROWE: I don't know if we can back up, but if you remember from the picture that was up earlier, the fuel element has two non-fuel plates running down beside the plates and then two other plates running this way. So the plates are inserted, and there is the seven-inch gap where there is no fuel. The top plate extends into that gap by a couple centimeters, but then it's just heavy water in them.

MEMBER ARMIJO: Overall, what's your fuel performance been like? Have you had fuel failures, leaking, swelling, bowing?

MR. ROWE: No.

MEMBER ARMIJO: I'm talking the more modern fuel.

MR. ROWE: Yes, and, again, I'm going to ask Tom to help me. I could answer, but I'd rather let him.

MR. MYERS: The only time a fuel element has failed was in the seventies. There was a pinhole leak. The staff at the time had a difficult time finding it. The only time you could see the leak was during operation. They shut down. They couldn't find it. They were able to locate the element simply by moving it to different positions. It was removed, and since then there have been no failures.

MEMBER ARMIJO: Okay.

MR. RICHARDS: The manufacturer of our fuel is BWXT out of Lynchburg. High temperature fuel element integrity. Maintaining the integrity of the fuel cladding requires that the cladding remain below the blistering temperature of 450 degrees C, 842 degrees F.

The way the tech specs have set up the LSSSs for the power not to exceed 130 percent full power, force flow 60 gpm per megawatt in the inner plenum, 235 gpm per megawatt in the outer plenum, and the reactor outlet temperature less than 147 degrees Fahrenheit.

These are the extreme conditions that we don't operate at these, but as long as we stay below these, there is no way that we could ever reach DNB, CHF, or OFI, and if we can't reach any of those, then there is no way we can reach the blistering temperature, so that is kind of very simply the way we have established the safety limit.

MEMBER ABDEL-KHALIK: But what is the most limiting phenomenon? I'm surprised that you're looking at OFI in this case.

MR. RICHARDS: What is the most limiting?

MEMBER ABDEL-KHALIK: Right.

MR. RICHARDS: The LSSSs would be the most limiting thing that we could run at. I'm correct with that, I think, Mike.

MR. ROWE: If I understand your question, are you asking what kind of an excursion would be most limiting?

MEMBER ABDEL-KHALIK: No. I'm asking whether DNB or OFI would be more limiting.

MR. ROWE: It depends on the flow regime.

MEMBER ABDEL-KHALIK: Pardon me?

MR. ROWE: It depends on which flow regime you're in. At different flows, different ones take over, so it's not -- it's not uniform, so we check against both of those cases, in all cases, but primarily when the reactor is operating, it is DNB. When the reactor is operating under normal conditions, DNB.

CHAIR SIEBER: On the other hand, when the fuel is just generating decay heat after shutdown, it can hang in the air without getting to 850. I read someplace in there. Is that correct?

MR. ROWE: Yes. When --

CHAIR SIEBER: That was a refueling accident where you lost the water.

MR. ROWE: Refueling accident where we --

CHAIR SIEBER: If you hang a fuel assembly in air --

MR. ROWE: Which we do, yes, after an appropriate cool-down, which is in the tech specs.

CHAIR SIEBER: Okay.

MR. ROWE: One hour for megawatt.

CHAIR SIEBER: Okay. You only get to about 800 degrees.

MR. ROWE: Yes, we do not -- the temperature will not rise. We use two sets of data to calculate that limit, to calculate that cool-down time, some data that were taken at Oak Ridge many years ago, where they did a very systematic study, and some data that were taken at this reactor when it was operating at 10 megawatts.

We took both those sets of data and extrapolated to our current operating conditions. We looked to find the hottest element, the one that would be most troubled by this, and we checked that element, if taken out and put in stagnant air, which is not actually what we do -- we actually put it in a helium environment, which would be better, but we look at stagnant air, and we remain below the blistering temperature under those conditions.

CHAIR SIEBER: Now, blistering in itself indicates the potential for fission gas release, as opposed to release of special nuclear material or non-gaseous fission products. What temperature would you have to achieve to have melting of the clad?

MR. ROWE: 650 degrees C.

CHAIR SIEBER: How much?

MR. ROWE: 650 degrees C.

CHAIR SIEBER: Centigrade?

MR. ROWE: Yes.

CHAIR SIEBER: Okay, and what is the alloy that's used, the aluminum alloy?

MR. ROWE: 6061, Aluminum 6061.

CHAIR SIEBER: 6061. Okay.

MR. ROWE: I'm not sure that that's the terminology people use nowadays, but it's the terminology I grew up with.

CHAIR SIEBER: Do you, by any chance, know what the alloying components of that are?

MR. ROWE: The major alloying components are magnesium and silicon, and that is done -- that is the hardening that is used. Those are the -- that's what you want to precipitate.

The 6061 alloy is set up so that there is slightly more silicon than is required to get the Mg_2Si that you used as a precipitating alloy, so you have a little bit more silicon in the alloy. That turns out to be very good when you start to look at embrittlement and hardening, which we likely will get to. In fact, I'm sure we'll get there.

CHAIR SIEBER: Yes. Well, maybe we ought to get to it now. The neutron capture by aluminum gives you a silicon isotope --

MR. ROWE: Correct.

CHAIR SIEBER: -- which, depending on the flux rate, would determine how rapidly the fuel assembly hardens or embrittles.

MR. ROWE: To be -- if I may, to just be careful about terminology, the alloy remains ductile.

CHAIR SIEBER: Okay.

MR. ROWE: Simply, it is a linear -- not linear. I should not say linear. It is a monotonic decrease in ductility with thermal neutron fluence --

CHAIR SIEBER: That's right.

MR. ROWE: -- which is alleviated, actually, by fast neutron fluence.

CHAIR SIEBER: Okay.

MR. ROWE: Opposite to what you expect, but it is true. It'll also make embrittlement in aluminum.

CHAIR SIEBER: Well, at Brookhaven there was some measurement of embrittlement in aluminum in some localized places, but you have -- the flux regime at Brookhaven is -- excuse me -- much higher than it is at --

MR. ROWE: In fact, we've made use of the Brookhaven data in analyzing our own situation, and so, yes, and it is true that at the end of this license extension which we're -- license renewal which we are requesting, we will not reach half of the fluence that we reached at Brookhaven.

MEMBER SHACK: But your thermal to fast ratio --

MR. ROWE: Very comparable in the region where they did measurements. We actually looked at that quite carefully to make sure that their measurements --

MEMBER SHACK: It is strongly dependent on that, or at least apparently strongly.

MR. ROWE: It is dependent. I would argue that it is not strongly dependent. It is dependent in the high fluence. In the high fast neutron fluence, you get less reduction in ductility for a given thermal neutron fluence.

MEMBER SHACK: What's your bounding thermal to fast ratio? Is it within the 21 that they have for the HFIR data to support Brookhaven?

MR. ROWE: Yes.

MEMBER SHACK: It's less than that?

MR. ROWE: In any region of high fluence and any region where you have a --

MEMBER SHACK: Okay.

MR. ROWE: -- have something happening, yes, that's correct. As I said, we did look rather carefully to be sure that the Brookhaven data could be used to represent our data.

MEMBER SHACK: But you never mention those ratios anywhere in the report that I could find.

MR. ROWE: In the SAR, I believe I did.

MEMBER SHACK: You did? Okay.

MR. ROWE: If you like, I'll dig up the reference for you, but I did put it in there.

CHAIR SIEBER: Well, I guess the concerns about embrittlement, if it exists at all to any degree, is does it change the properties of any pressure-retaining in the neutron fluence at your outer wall or the container that holds, you know, the deuterium oxide is probably pretty low.

MR. ROWE: It most assuredly is. The highest fluence rate that we have is, of course, the beam tube tips.

CHAIR SIEBER: Right.

MR. ROWE: But the stress there is --

CHAIR SIEBER: Brookhaven found --

MR. ROWE: The stress is minimal there.

CHAIR SIEBER: -- where that was not a problem.

MR. ROWE: In fact, the stress is compressive, not tensile, and the ductility remains at 80 percent. At the end of this license renewal that we are requesting, we'll still have better than 80 percent of the original ductility.

I did also do an analysis of chart the impact strength and years to leak-before-break criterion, and I took the worst case of radiation, which is the beam tube tip, and the highest stress, which is out at the boundary. The two could never happen in the same place.

CHAIR SIEBER: Okay.

MR. ROWE: Nonetheless, that's what I did as a calculation, and we satisfied leak-before-break criterion --

CHAIR SIEBER: Okay.

MR. ROWE: -- well beyond the end of this renewal. If anybody wants to renew after this, they will have to redo the arguments and convince your successor, but we have done that calculation.

MEMBER ARMIJO: At some point, you know, has the staff reviewed that? I guess that's appropriate to ask the staff, you know, that calculation to see if it makes sense to you guys. You'll get to that?

MR. KENNEDY: Yes.

MEMBER ARMIJO: Okay.

CHAIR SIEBER: I was going to say the court reporter would have to indicate that your head was going up and down. I actually don't have any further questions on that, but if any other member does --

MEMBER ARMIJO: Yes, at some point I'm still concerned about all the spent fuel. Is there a segment in this presentation where you talk about where it goes and how it's cooled and protected and all those kinds of things?

MR. RICHARDS: We hadn't -- it's not in this presentation, but if you want to talk about that, we can certainly do that.

MEMBER ARMIJO: It's up to the Chairman when it makes sense. See, there's an awful lot of fuel out there.

CHAIR SIEBER: I think, and the staff can please correct me if I'm wrong, but the subject matter of the license renewal involves the handling of the fuel while it's on the site as opposed to where it goes and how it goes, because the casks are licensed individually --

MEMBER ARMIJO: My question is just related to the -- Jack.

CHAIR SIEBER: -- and so is Savannah River.

MEMBER ARMIJO: Jack, my question is just limited to what happens on site --

CHAIR SIEBER: Okay.

MEMBER ARMIJO: -- you know, how much fuel is there, how is it cooled, how is it protected, whatever.

MR. RICHARDS: You know, we have to be a little careful, because we can get into security issues here.

MEMBER ARMIJO: I understand, so just tell me when that's --

MR. RICHARDS: Yes.

CHAIR SIEBER: Well, we do know that it can hang in air and not melt.

MR. RICHARDS: Yes.

CHAIR SIEBER: Okay. So air cooling is good enough, even though from a hazard standpoint you probably -- the health physicists probably wouldn't care for that.

MEMBER ABDEL-KHALIK: I have a different kind of question. The numbers you have on there with regard to the average heat flux, the peaking factor, and the minimum critical heat flux ratio imply that your critical heat flux is in the neighborhood of 4.6 megawatts per square meter. Where does that number come from?

MR. ROWE: This is based on -- maybe, Bob, you want to answer that, or I can. It doesn't matter. I mean, it is based on looking at different critical heat flux correlations.

MEMBER ABDEL-KHALIK: And there are correlations available for the range of operating conditions that you expect?

MR. ROWE: Yes, there are at least three or four different ones. What we've tried to do is to look at the one that had the most experimental verification, and we continue to look at them all the time, as everybody else does. So, yes, there are measurements which have been done.

The correlation that was used in the SAR is the one due to Mirshak. That was a correlation. It was derived for plate-type fuel. It was, in fact, aimed at this kind of fuel. It has the right pressures, the right flow velocities. In addition, there have been later studies, which we have also taken advantage of and looked at and compared to. We believe that using the Mirshak correlation is, in fact, conservative compared to other correlations that we could have used.

MEMBER ABDEL-KHALIK: I guess the number seems a little -- the critical heat flux number seems a bit higher than what I would have expected for such low pressure conditions, but --

MR. ROWE: As I say, I don't know whether you want me to take the time now, but we could go through it. It was done, as I say, using more than one correlation. In fact, we compared several of them. We continue to compare them. As you well know, people are still doing measurements, so you'll always have a new one to look at, and you always have new data to look at, primarily not so much for reactors anymore as for spallation sources where they have very high velocities and narrow channels.

CHAIR SIEBER: I guess if there are no other questions, we're -- the time is moving rapidly, so if we could just briefly go --

MR. RICHARDS: I'll try to go --

CHAIR SIEBER: -- through as much as you can.

MR. RICHARDS: The engineered safety features, emergency power, we have the -- under these categories we have the emergency core cooling system, 3,000-gallon emergency core cooling tank we have that's approximately 37 feet above the core. There is sufficient $D_2 0$ in the emergency cooling tank to provide about two and one-half hours of cooling on a once-through basis. The $D_2 0$ from the 14,000-gallon $D_2 0$ storage tank could also be used to be pumped back up into the tank and down through the core.

The two and one-half hours is to allow us to -- if we have to, we can bring in other sources of cooling for the core. We have redundant building feeders. We have two diesel generators. We have the station batteries in case we should lose the on-site power.

MEMBER RAY: Okay. I know the Chairman needs us to move along here, but I couldn't understand the description of the emergency power and the loss of off-site power sequence and all that sort of thing, so let me see if I can just ask several questions. It said the diesel generators -- apparently, you can continue power reactor operations indefinitely with one diesel generator out of service.

MR. RICHARDS: We only --

MEMBER RAY: That's what I read.

MR. RICHARDS: Yes.

MEMBER RAY: You just need one?

MR. RICHARDS: Yes.

MEMBER RAY: Okay. Are these things train-aligned? In other words, is there some set of stuff that isn't serviceable when one of the diesel generators is out of service?

MR. RICHARDS: I'll have to defer to Tom. He's the expert on that.

MR. MYERS: No.

MEMBER RAY: So the two diesel generators are just redundantly supplying a single set of engineered safety feature power supplies?

MR. MYERS: That's correct. In fact, the diesels are actually not necessary. The battery is sufficient to keep the reactor safe in a shut-down condition.

MEMBER RAY: Well, you test the diesel once a month.

MR. MYERS: Yes.

MEMBER RAY: Why do you do that if it's not necessary?

MR. MYERS: Just for going the extra mile. We're providing an extra redundancy.

MEMBER RAY: John?

MEMBER STETKAR: You only have one battery and one DC bus, correct?

MR. MYERS: Yes.

MEMBER STETKAR: So have you looked at what are the consequences, for example, if you have a catastrophic failure of that DC bus, short to ground? You know, I'm not going to postulate how the failure occurs. They have occurred. What happens in the plant if you lose that DC bus?

MR. ROWE: We analyzed the accident of what happens if we have no power following a loss of off-site power. We have no power in the shut-down pumps.

MEMBER STETKAR: That's AC power. I'm asking about DC power.

MR. ROWE: We have no power of any kind. The accident we analyzed was that the shut-down pumps did not come on, and that was not -- that did not --

MEMBER STETKAR: I'm still asking about DC power, because DC power also supplies instrumentation and control signals to valves and things like that.

MR. ROWE: We would have -- we would lose, of course -- you're quite right -- all of that indication, but, in fact, so long as the fuel remains covered, as long as we have water around the fuel, the reactor is safe.

MEMBER STETKAR: Let me ask you then, because we're on this slide, you mentioned the D_2O emergency cooling tank. The outlet from that tank, there are four pneumatically operated, normally closed valves that must open. Can those valves open if you have a) no air pressure, or b) no DC power? Can those valves open?

MR. ROWE: Let me give it a shot, and then Tom will correct me if I make a mistake, or, Tom, you can go ahead if you wish. The initial two and one-half hours, nothing happened. That is, gravity only has to continue working, and that is the tank that is in the top of the reactor. It's always maintained full. When you're putting water in it, it keeps running out.

MEMBER STETKAR: But the valves that allow the water to come out of that tank are normally closed, aren't they?

MR. MYERS: Yes, they are, but they can be manually opened, as well.

MEMBER STETKAR: Manually and mechanically opened?

MR. MYERS: They are in the vicinity of the control room. They're within about 20 seconds.

MEMBER STETKAR: That's the answer I was looking for. Thanks. As long as they can be manually and mechanically opened --

MR. ROWE: And we do have time, because the initial feed of water is instantaneous without --

MEMBER STETKAR: You have the 800 gallons.

MR. ROWE: That's right. That gives us about half an hour.

CHAIR SIEBER: When you lose electrical power, everything goes to its safe conditions, the shim rods inserted.

MR. ROWE: The shim rods will be inserted.

CHAIR SIEBER: Okay. Go ahead.

MR. RICHARDS: Major modifications since the last renewal. Tom mentioned before that we had some tech spec administrative changes, confinement building, penetration isolations for the guide hall. Plate and frame heat exchangers were changed out. Cryostat was installed. The nuclear instrumentation was replaced. Fuel loading and type -- of course, we went from 300 to 350 grams. Switchboards, batteries, and UPSs have just been updated, and the plume abatement tower -- in 2005, we put in a new plume abatement tower.

CHAIR SIEBER: How often would you change your station battery?

MR. RICHARDS: That just came up a couple days ago. You change your batteries every two years, three years.

MR. MYERS: Are you referring to station battery?

CHAIR SIEBER: Yes.

MR. MYERS: Yes, the station battery cells are checked on a regular basis, tech spec requirement. We did change out the station battery in the last five years.

CHAIR SIEBER: Okay. So you rely on your surveillance tests to determine what capacity you have left in the battery?

MR. MYERS: Yes.

CHAIR SIEBER: Okay. Thank you.

MR. RICHARDS: Next item was the physics, reactor physics parameters. We've listed our temperature coefficients, moderator void coefficients, fuel temperature coefficient. You might note that prompt neutron lifetime is pretty long. We've got a

D_2O reactor here, and the delayed neutron fraction is also another number that you're probably not used to seeing numbers quite that big, small.

MEMBER ABDEL-KHALIK: Is this moderator void coefficient number correct?

MR. RICHARDS: Moderator -- the void coefficient?

MEMBER ABDEL-KHALIK: Right.

MR. RICHARDS: Bob?

MR. WILLIAMS: Bob Williams, nuclear engineer. That number is the smallest number. The moderator -- the void coefficient and the temperature coefficient sort of depend on where the moderator is located, because we have a pretty wide gap between our fuel elements, so that's the smallest value.

MEMBER ABDEL-KHALIK: Okay, it looked awfully small.

MR. WILLIAMS: Which one?

MEMBER ABDEL-KHALIK: The -.03 --

MR. WILLIAMS: The smallest value really depends where the moderator is, and I've looked all over. It's always been negative. We do have a lot of moderator outside our fuel elements.

MR. RICHARDS: There is a very large spacing, too. It's rather unusual.

CHAIR SIEBER: Okay.

MR. RICHARDS: Credible accidents. We have analyzed in our SAR the reactivity insertion accident, loss of primary coolant accident, the loss of flow, improper fuel handling. None of these result in any fuel damage. This is the blistering temperature, of course.

MEMBER ARMIJO: I'm sorry. In the reactivity insertion, do you include cold moderator coming in, you know, a big slug of cold D_2O coming into the reactor? Is that one of the events that you analyzed?

MR. RICHARDS: I think we started --

MR. ROWE: No, that is one that we looked at and did not analyze, because we did not see a credible mechanism to get a large slug suddenly in. We're continuously flowing from the heat extender. The reactivity -- I'm sorry.

CHAIR SIEBER: You are already cold.

MEMBER ARMIJO: Hundred degrees.

CHAIR SIEBER: Hundred degrees.

MR. RICHARDS: Well, we always flow the primary, though, so we don't let it get cold.

MR. ROWE: We are in continuous circulation, so we're always coming from the basin.

There is no way to change the basin temperature suddenly. The basin is a large reservoir of D_2O, and you simply can't make a rapid change in that, so, as I said, we didn't do it. The accidents we did analyze in detail were the start-up accident and removal of the most reactive experiment allowed by our tech spec, the rapid removal, the most reactive experiment that is allowed in our tech specs, and so those are the two we did look at.

MEMBER MAYNARD: Is there any credible way for an inadvertent opening of the valve that allowed the gravity feed to get cooler water in there, the water that's up above?

MR. ROWE: The water up above? I would say no, but I'm going to ask Tom to comment, as well. I would rather he answer.

MEMBER ARMIJO: That's not heated, right? That's stored at ambient --

MR. ROWE: It's stored at the temperature inside, which is rarely very cool.

MR. MYERS: It's important to note that the -- I think you're referring to the emergency tank --

MEMBER ARMIJO: Yes.

MR. MYERS: -- the 3,000-gallon tank. The 3,000-gallon tank is continuously recirculated with the primary system. Water is pumped up from a storage tank to the emergency tank. It overflows to the vessel, overflows to the storage tank, so while it's cooler than the vessel, it's going to be in the neighborhood of 80 degrees Fahrenheit.

MEMBER ARMIJO: Okay.

MEMBER RYAN: Just a general question on credible accidents. Have you analyzed intentional acts?

MR. RICHARDS: No, we have not.

MR. ROWE: Not really, no. I mean, in some sense, you can say the start-up accident could be considered that way, but other than --

MEMBER RYAN: No, I'm thinking of somebody that wants to do something bad and has intent to do that.

MR. ROWE: I'm not sure how --

MEMBER RYAN: I can't think of a lot of detail, but that's --

MR. ROWE: -- far I can go into that --

MEMBER RYAN: -- analysis at this point.

MR. ROWE: -- unless we --

CHAIR SIEBER: We would have to close the meeting if there is such an analysis.

MR. RICHARDS: It's not something we're not aware of.

MR. ROWE: We've thought about is, I guess, as much as I can say.

MEMBER STETKAR: I had a question about your loss of flow accidents. I read through all of them. There seems to be one -- there is a common outlet valve from the reactor, the suction side to the primary coolant pumps. It's DMV 19. I didn't see an analysis about what would happen if that valve closes spuriously.

You looked at spurious closure individually of each of the two inlet valves, but I didn't see an analysis about what would happen if that valve closes. That would seem to shut off all flow to the core, and it also isolates the relief valve from the reactor vessel, because the relief valve is on the suction header side of that valve.

So I was curious about why you haven't looked at closure of that valve, in other words, catastrophic loss of all flow through the core. Have you looked at that?

MR. ROWE: We did not analyze that accident.

MEMBER STETKAR: Why?

MR. ROWE: Well, we didn't -- I have to say we did not analyze it, because we did not look on it as a bigger challenge.

MEMBER STETKAR: You may want to.

MR. ROWE: But I will think about it while we're in here, and maybe when we break we'll come back here --

MEMBER STETKAR: Great. Thanks.

MR. ROWE: -- with an answer. I'll think back to our reasoning at the time.

MEMBER STETKAR: Thank you.

CHAIR SIEBER: Thinking about the break, maybe we should move on.

MEMBER SHACK: Well, I was just curious here on the reactivity insertion accident where you found that the RELAP point kinetics model incorrectly predicted the power excursion, so you whipped out a solution with Netlab. Is this a basic problem with RELAP?

MR. ROWE: That has been corrected. Yes, it was, and yes, it has been corrected. The code has been changed. If you want details, again, I'm happy to provide them at the break about exactly what was wrong.

MEMBER SHACK: Okay. I'm just surprised that it took all these years of calculations.

CHAIR SIEBER: So was I.

MR. RICHARDS: All right. The maximum hypothetical accident assumes a complete flow blockage of one of the elements, and I think, as you pointed out, we don't know of any credible way to do this, but this is the accident that we did analyze in complete, instantaneous melting. The bottom line is that we still don't end up with any excessive doses at the site boundary. We have a 400-meter site boundary onsite, so --

MEMBER RAY: But the problem with that is, you know, it's hard for us to look at an accident like that, recognizing it's hypothetical, and then assume that everything works and ask what the dose is, because there is no -- at least, I have no way to discern any redundancy anywhere in the stuff that's required to work, and so we're just not used to thinking about a world in which everything works the way it's supposed to.

MR. RICHARDS: I think the regulations say we have to --

MEMBER RAY: I don't -- I'm not questioning the regulations. I'm just -- we go through the exercise, but what meaning to give to it is hard to say, because most of us with experience running plants come to believe that not everything works all the time. You have to basically go back and say, "But this will never happen."

MR. RICHARDS: Well, I think that's what we said.

MEMBER RAY: I understand.

MR. RICHARDS: There is no credible way for this to happen.

CHAIR SIEBER: In, fact, a lot of -- if you think about it, a lot of things have to fail in order to even get close to this kind of an accident.

MEMBER RAY: Well, that's what I mean, Jack. The hypothetical is what makes it okay. It's just not possible.

CHAIR SIEBER: I think what they're trying to demonstrate is if everything failed, you're still going to be under Part 20.

MEMBER RAY: Well, everything except all the engineering and safety features. They all have to work.

CHAIR SIEBER: The building has to work.

MEMBER RAY: Yes, radiation detectors and the isolation capability.

CHAIR SIEBER: Radiation detectors, you know, they don't have to work.

MEMBER RYAN: Well, in this case, I think --

CHAIR SIEBER: If you have an accident, you know, you take what's there.

MEMBER RYAN: In this case, the immediate release of the entire fission product inventory to the reactor vessel, so the reactor vessel is doing something in the containment building, but --

CHAIR SIEBER: They have to maintain integrity.

MEMBER RYAN: What?

CHAIR SIEBER: They have to maintain their integrity.

MEMBER RYAN: Right.

MEMBER RAY: I'm not sure what the entire sequence looks like. I'm just observing that nothing fails once you hypothesize the condition. Everything works.

MR. ROWE: But, if I may, just a comment. It was a very conservatively chosen MHA was the way to release a very large amount of radiation. We don't have any mechanism to actually do this. I understand your -- I understand your reservation, but, in fact, we have assumed -- in assuming the accident, we have assumed already that one of our passive features failed, which is a screen that wouldn't let a blocking piece that big get there.

MEMBER RAY: Well, I know, and Jack would say it's more than one thing that's failed to get you there in the first place.

MR. ROWE: I understand your concern.

MEMBER RAY: It's just a matter that to then say everything works the way it should, and look at what a low dose we get --

CHAIR SIEBER: Well, things would have to work or the emergency ventilation system in the integrity of the building.

MR. ROWE: Confinement.

MEMBER RAY: Right.

MR. ROWE: That's what we assume, the emergency ventilation and confinement.

CHAIR SIEBER: Yes, why don't we -- if you don't have -- if there are no more questions, why don't we move on?

MR. RICHARDS: This is more of the MHA, just more of the details of the MHA. Here are the actual doses at the 400-meter boundary. I think this was -- I think Bill mentioned this, 7MR for the operations.

MEMBER RYAN: I'm assuming you used very

negative meteorological conditions to bring your radioactive material down onsite?

MR. RICHARDS: These are the worst conditions.

MEMBER RYAN: A tornado, something like that?

MR. RICHARDS: Yes.

MEMBER RYAN: Yes.

MR. RICHARDS: This is just a slide showing the ALARA that we have at the facility. These are the operational doses for the various years. In 2001, we did an extremely hot job in the thermal column, and that's why you see that piece there. The number of people badged divided by the -- well, the dose divided by the number of people badged gives you the --

MEMBER RYAN: Could you give us --

MR. RICHARDS: -- total equivalent dose.

MEMBER RYAN: I'm sorry. Could you give us any insights into -- you know, I appreciate these curves of the data, but it doesn't tell me how well did you do versus how well did you plan it. I mean, one hot job, I mean, that's maybe okay, because if you planned it that way and it turned out the way you planned it, that's okay from an ALARA standpoint.

I'm trying to get an insight as to how did

ALARA play into these doses. Were you satisfied with them year by year? And, I guess, being an experimental facility, you'd expect annual ups and downs, but from a health physics point of view I'd be thinking about, "Did it turn out the way we thought it would, or was it higher or lower or what?"

MR. RICHARDS: I'll let Dave -- he's the one that -- I can tell you that we have radiation work permits, and jobs are planned, but I'll let Dave speak to that.

MR. BROWN: Dave Brown, Health Physics. In fact, the doses you see there in 2001-2002 were multiple large-dose projects, not just one, and I think in every case we met our ALARA goals.

MR. ROWE: Just if I can add, we did do an ALARA review at the end of each of those projects, a formal -- we got together and went through from an ALARA perspective how well we had done. We did ALARA planning before we did them, and we did an ALARA review at the end.

MEMBER RYAN: Did any of those reviews result in significant operational changes for subsequent experiments?

MR. BROWN: Nothing significant, no.

MEMBER RYAN: No?

MR. BROWN: We never really ran into anything we didn't expect in those cases.

MEMBER RYAN: Got it. Okay.

MR. RICHARDS: Baseline material. This has more to do with the reactor vessel. I think we've talked a little bit about the effects of ductility. We did a visual inspection of the vessel. I've gotten out of sequence here. A visual inspection was done in 2004, and an ultrasonic testing of the primary was done in 2001, so we do have a program --

MEMBER SHACK: But that was just back-wall reflection. You weren't looking for cracks.

MR. RICHARDS: That was just back-wall reflection, yes, thickness. The surveillance program we have in place for the aging surveillance, actually, follows the ANS-15.1. This is the development of technical specification for research reactor Section 4.

That section is surveillance requirements, and it specifies the frequency and scope of surveillance to demonstrate the meaning of performance levels for reactor systems that are safety related. It's also an NRC-endorsed standard, and that's the one we follow, and, of course, we have the management commitment to, through our budgets and spending plans, to make sure that we continuously upgrade the systems of the reactor.

CHAIR SIEBER: And that's contained in your technical specifications?

MR. RICHARDS: Yes, sir, it is.

CHAIR SIEBER: Okay, which is this document here.

MEMBER ARMIJO: In the RAIs, the response to the RAIs, you mentioned or you stated that the vessel was last inspected in 2003 by -- and apparently by a visual technique.

MEMBER SHACK: Yes, binoculars.

MEMBER ARMIJO: Yes. No, it wasn't.

MEMBER SHACK: Someone said binoculars.

MEMBER ARMIJO: I heard that too, but I'm not sure. I want to ask. Exactly how was that -- how good was that visual inspection?

MR. RICHARDS: It was very good. We did it with -- Tom, do you want to talk about that inspection? His crew did it.

MR. MYERS: Yes, there were several inspections in 2001-2002, maybe one in 2003, but in 2004 we obtained some pretty good camera equipment for high-rad equipment, and so we did the inspection in September of 2004.

We have different areas we can insert the camera equipment into the core and look at the beam tips specifically, as well as the rabbit tips, which are the pneumatic thimbles in the vessel, and we did a thorough inspection of everything. We recorded it, and it's got good resolution.

MR. MYERS: Okay.

CHAIR SIEBER: Did you note any deterioration that required any kind of repair or accelerated surveillance?

MR. MYERS: We did not.

MEMBER ARMIJO: Nothing unexpected. There was no pitting? There was no physical damage?

MR. RICHARDS: We didn't observe anything that would require attention.

MEMBER ARMIJO: That's something that had been operating since 19-whatever.

CHAIR SIEBER: `62 or whatever.

MR. RICHARDS: Okay. That is it.

CHAIR SIEBER: Okay. Right on time. Any questions?

MEMBER RAY: I couldn't figure out where to ask the question, so just tell me if it's going to come up later. In looking at the environmental monitoring, I notice the effluent pathways are monitored and so on, as you expect, but is there any groundwater monitoring wells?

MR. ROWE: Yes. Do you want to have Dave do it? Dave is the one that does the monitoring, so he can --

MEMBER RAY: Yes, just looking for unmonitored pathway sampling to determine if there is leakage into the groundwater from like a spent fuel storage area or something of that kind.

MR. BROWN: We have currently very limited groundwater monitoring. We have two locations that we sample. One is onsite, and it's upstream, and one residential well downstream, which we have access to most of the year. They do winterize, so we don't get access to it during the winter months.

MEMBER RAY: Do you feel this is sufficient to detect any or to recognize any undetected leakage into the groundwater or not? Okay.

MR. BROWN: That's a pretty open question. We have no indication of a leakage into the groundwater, either from monitoring our systems in-house -- you know, we don't see the level of our storage pool decreasing unexpectedly, and the wells and streams that we do sample, we see no positive indication of tritium, which is our primary indicator.

MEMBER RAY: Right. Well, it's hard to separate a small leakage into the ground from evaporation of a pool, for example, so it's hard to reach any conclusion just from that makeup.

MR. BROWN: We have no wells in close proximity to the building, so.

MEMBER RAY: Okay.

CHAIR SIEBER: Any further questions? If not, we are a few minutes late, but I would still like to return by that clock at 10:20. Thank you very much.

(Whereupon, the foregoing matter went off the record at 10:03 a.m. and resumed at 10:19 a.m.)

CHAIR SIEBER: Okay. Yes, sir.

MR. ROWE: Thank you, Mr. Chairman. I consulted with my colleagues, and we did think about this question, although not exactly in the way you phrased it, and we have not done a full analysis, but what would happen is the stroke time on that valve -- it's a motorized valve. It is not an air-operated valve. The stroke time is 21 seconds, and the flow would decrease over 21 seconds.

When that valve was completely closed, we would maintain natural circulation within the vessel. It would be set up. We have done that calculation.

There are pathways.

MEMBER STETKAR: Where is the heat -- where is the heat sink, just to the vessel walls?

MR. ROWE: Just to the thermal -- to the biological shield. So, as I said, we have not done the detailed calculation, and I don't want to misrepresent what I'm saying, but we have thought about the issue of what would happen if the vessel became isolated.

MEMBER STETKAR: That wouldn't account for if the -- I don't know what kind of -- thanks.

MR. ROWE: Anyway --

MEMBER STETKAR: We're going to keep on moving. Okay.

MR. ROWE: That's just a -- I said I would --

MEMBER STETKAR: Thanks.

CHAIR SIEBER: If you have additional questions, go ahead.

MEMBER STETKAR: Yes, but it gets into a lot of detail. We should probably -- if there is time left over at the end, Jack, we might bring it back up.

CHAIR SIEBER: Yes, we'll have a discussion at the end.

MEMBER STETKAR: That's fine.

CHAIR SIEBER: Okay. I'd like to invite the staff. Bill? And this is the second set of slides that we got.

MR. KENNEDY: All right. In this portion of the presentation, I am going to be going through these topics with the exception of the staff inspection history, which Johnny Eads will present. I'll start with an overview of our Safety Evaluation Report, results of the application of our SRP, and major issues. There's actually really one major issue besides the open item and also our principal safety conclusions.

As I mentioned, we did conduct our review using our Standard Review Plan, NUREG-1537. We looked at a variety of information sources when we did our review. We looked at, of course, the renewal application, including the safety analysis report, the technical specifications, both as they were submitted with the original application and a revised set of technical specifications that we received about halfway through the review.

We also looked at the responses to the staff's REIs, and we looked at the annual reports. I looked at annual --

MEMBER RAY: Excuse me.

MR. KENNEDY: Yes?

MEMBER RAY: Could you just mention why there was a revised set of tech specs you received at that point?

MR. KENNEDY: As a result of the REIs, we asked -- it kind of became apparent that an update of the tech specs would really be helpful as part of the renewal, and so with the response to the first set of REIs, I believe NIST did submit a revised set of technical specifications. I'll cover that in more detail a little bit later.

Also, we looked at -- we used first-hand observations from actually going out to the facility for site visits and discussions and some use of our inspection reports, as well. The staff -- the NRC staff also used technical evaluation input on the SAR, the tech specs, and the REI responses that was provided under contract with the NRC by Washington Safety Management Solutions.

So, following our SRP and the established NRC procedures for researching test reactor license renewal, the Safety Evaluation Report covers a similar range of areas that would be reviewed in the initial application for this type of a reactor or for initial facility license.

Areas of review, I'll go through these quickly, but they include site characteristics, which would be hydrology, seismology, the design of the control elements and the core support structure, the nuclear design for control of the reactor and coefficients, the moderator.

We looked at the design of the primary system and the secondary cooling systems, the confinement building, the ventilation systems that would be included in the engineering safety features, the reactor control system, the reactor protection system, and also the radiation monitoring system. That would be under instrumentation and controls.

We looked at normal power and backup electrical power. Normal ventilation falls under auxiliary systems, and I believe there was a question earlier about spent fuel handling and storage and what happens to it onsite, and that's covered under our auxiliary systems, where we actually look at the spent fuel pool and transportation of the fuel from the reactor vessel to that pool.

Also looked at the, you know, experimental programs that are going on, and that includes administrative controls of experimental programs, which was another issue that had been raised, radiation protection, waste management.

Also looked at how the licensee conducts their operations, and a lot of this is contained in Section 6 of the technical specifications that covers their internal review committees, the staffing of the reactor, the reporting and record keeping.

Kind of in the peripheral reviews to this main review, we also reviewed emergency planning, security planning, and the operator training and requalification program, and as I mentioned before, one of the open -- the only open item is in regard to the operator training and requalification plan.

We looked at a maximum hypothetical accident as a bounding accident for all credible accidents. We looked at the credible accidents.

A lot of the review was focused on making sure that the tech specs were adequate and up-to-date. We also looked at decommissioning planning from the respect of ensuring that there was going to be enough money to decommission the reactor. We didn't do a detailed review of an actual decommissioning plan, because they're not required to submit that to us at this time, but we did do some review of decommissioning planning, and we looked at prior use of reactor components.

CHAIR SIEBER: I might point out that the reactor is actually owned by the federal government and administered through the Department of Commerce, and so to my knowledge there has not been money appropriated to decommission this, but in today's age, I presume that Congress would act appropriately.

MEMBER ARMIJO: If they don't have the money, we've got problems.

CHAIR SIEBER: Yes, this is my personal feeling.

MR. KENNEDY: Okay. I am going to talk about application of our Standard Review Plan. We applied NUREG-1537 during our review, including all of the supporting guidance and documents that I had mentioned before, the ANSI standards, the reg guides, and the NUREGs.

So, because this SRP was written to cover all non-power reactor licensing actions, we had to go ahead and, you know, apply only the portions that were really applicable to this unique case and that we actually thought were within the scope of the license renewal. So the series of the next 15 or so slides is more detailed coverage of all the areas that we reviewed, so if there are areas you would like to focus on or skip, again, you can just let me know.

So, for our siting criteria we did some review of the geography just in terms of what's around the facility. What's the topography like? Were there any unique -- any unique considerations in terms of effluent release concentrations or accident releases?

We looked at the demography of the area and population growth, looked at whether or not there were any nearby facilities or transportation industries that could potentially impact the reactor in case of an accident at one of those facilities. We did review the meteorology of the site and the hydrology and also did --

MEMBER RAY: In that regard --

MR. KENNEDY: Yes?

MEMBER RAY: -- a question you've probably heard me ask about ground well monitoring. The hydrology you describe is all surface hydrology. What about sub-surface and the possibility of unmonitored pathways leading to sub-surface acquifers? Do you think the monitoring is adequate?

MR. KENNEDY: Well, there was a monitoring program that had been in place for 20 or so years that involved a lot of wells, and I guess that program had been discontinued. It was my understanding that some of the surface monitoring that they did was also groundwater monitoring in the respect that the water table there, the depth below the surface at which the groundwater resides, is pretty shallow and that that would actually show up in one of the ponds that they do sample that is southwest of the site, which is the direction of the groundwater flow.

CHAIR SIEBER: Is that pond within the controlled area?

MR. KENNEDY: Yes.

CHAIR SIEBER: I think it is from the map that I saw.

MR. KENNEDY: Yes, it is.

MEMBER RYAN: It sounds like all these flows have been established after the reactor was built over a long period of time. Is that a fair statement?

MR. KENNEDY: I believe that's a fair statement.

MEMBER RYAN: Okay. All right.

MR. KENNEDY: I'm relying on data and information provided by the licensee in their environmental report that was submitted with the renewal application.

MEMBER RYAN: Okay.

MR. KENNEDY: And that had pretty extensive discussion of groundwater flows, directions, and velocities and potential for capture of radionuclides by the soil based on whether it's carbon content.

MEMBER RYAN: Well, with tritium being of interest being at power reactors these days, I think that's certainly what's in my mind and perhaps what Mr. Ray is thinking about, as well, is that tritium is usually a leading indicator. Without too much trouble you can sample in a few key spots to make sure that's the case.

I think I'm just probing as do you feel comfortable that is the case, that the environmental monitoring that is done is capable of being a leading indicator for any issue of tritium coming up, whether it's coming out of the stack and back down in the ground or from the facility itself.

CHAIR SIEBER: Yes, but the presence of carbonaceous trash, ordinarily that's a ion exchanger and concentrates radioactivity, but with tritium it just moves right on through, in fact, so you have to look for pretty low levels. You have to be able to detect pretty low levels.

MEMBER RYAN: It's pretty easy to get down to 400 picocuries per liter, I mean, which is a tiny fraction of -- well, it is background in a lot of areas but below background in some.

MR. KENNEDY: Well, I'll mention when I talk about the fuel pool, as well, the water in that fuel pool is not below the regulatory limits for uncontrolled release in terms of the tritium concentration, but it is perhaps 10 to 200 times the release concentration limit.

Also, my understanding is that the fuel pool is at such a depth that the hydrostatic pressure is actually into the pool such that if there was degradation of the liner or any cracking, they would actually -- the licensee would pick up on their fuel pool conductivity monitoring a spike in conductivity due to impure groundwater leaking into the fuel pool.

MEMBER RYAN: One measurement culminates a lot of questions.

MEMBER ABDEL-KHALIK: Did I hear you say earlier that there was a much more extensive well monitoring program in the past that was discontinued?

MR. KENNEDY: That's my understanding. I guess I would defer to the licensee if they would like to --

MR. RICHARDS: There have been a lot of the wells that we were monitoring 20 years ago that had been closed. That's correct, isn't it, Dave?

MR. BROWN: That is correct, and I am not sure if they were all residential wells or not, but all the residential wells pretty much in the area have closed, because they've gone go public water. In the original siting of the reactor, I believe there was a monitoring program that involved digging some wells, but I don't have any details on that.

MEMBER STETKAR: I have two questions, and it's on the meteorological side of this issue. One is that the confinement building is apparently designed for 100 mile-an-hour wind speed, and there are calculations in the SAR, and I recognize that this is not a probabilistic analysis, but people throw around numbers like the 100-year maximum wind speed, which in effect is a probabilistic type of calculation, and the calculations in the SAR extrapolate from an ASCE standard 50 mile-per-hour or 50-year maximum return period wind speed of 90 miles per hour to a 100-year maximum wind speed by using two very, very precise numerical factors, which, as far as I can trace them, are derived from Caribbean hurricane data, and statements are made that, "Well, in addition to that, in 39 years we've never observed a wind speed higher than 55 miles per hour at Dulles."

So I went back to the meteorological data, and I found three times last year where wind speeds at Dulles and Reagan were 60 miles an hour, 66 miles an hour, and 74 miles an hour. So I was wondering, first of all, what is the basis for those very, very precise numerical scaling factors that determine that the 100-year return period maximum wind speed miraculously is 100 miles per hour, and two, what analyses have been done of the real historical wind speed data here to look at the likelihood of exceeding that 100 mile-per-hour peak gust wind speed, which is the thing we're interested in, not average sustained wind speed?

There were several RAIs on this topic, and apparently the staff satisfied itself that all of these calculations were acceptable. Is that correct?

MR. KENNEDY: Yes, we satisfied ourselves that these were reasonable calculations.

MEMBER STETKAR: All right.

MR. KENNEDY: To give you as detailed an answer as your question, I'm not prepared to do that right now.

MEMBER STETKAR: I recognize that, but that might be a takeaway for something to think about for the full committee presentation or in later discussions, because there was some mystery in there to me. The second -- the second question -- I'll let you finish writing your notes there, so. Go on.

The second question is on the snow loading on the roof, the roof is designed for 25 psi load, and the criteria require a maximum of snow loading plus a rain-on-snow loading, and the rain-on-snow loading analyses were done assuming that the historical precipitation in the months of -- I think it was December through March but essentially the winter months was essentially 50 percent rain and 50 percent snow, and with that assumption the loading on the roof, the rain over the snow, came out to be just less than the -- came out to be less than the 25 psi maximum.

If you assume 60 percent rain and 40 percent snow, you hit the 25 percent, the 25 psi. If I look back at the actual weather conditions, there are several events where the actual precipitation that occurred during January and February was a couple inches' worth of rain, and indeed if there had been a snow loading, it would have exceeded the 25 pounds.

So here is, again, I'm not sure what's the basis for that 50 percent assumption, because that 50 percent assumption is the key to why you meet the 25-pound loading criteria, and there were a couple of questions about that, also, but I was curious how the staff satisfied themselves that the calculations and that assumption was justified. That may be also, I recognize, pretty detailed, but write it down perhaps as a takeaway for --

MEMBER RAY: Well, John, less detail but on the first point that you said, I was puzzled. Maybe you know the answer to this. It says, "The confinement structure is designed for 100 mile-an-hour wind load, which is within the uncertainty for the 100-year return period wind load," which sounds a little like the question you were asking, but it was --

MEMBER STETKAR: I didn't get into it. The actual calculation that they did showed that the 100-year return period maximum wind speed was 102.5 miles per hour, and they argued that that was close enough to 100 miles that it wasn't a problem. So I think that was the source of the uncertainty, but that raised the flag to me about how was that calculation actually performed and led to the more detailed question.

MR. KENNEDY: Okay. I will take that away from this meeting and address your concern.

MEMBER STETKAR: Thank you.

CHAIR SIEBER: I would expect in our final meeting that you could have the answers for those.

MR. KENNEDY: Yes.

CHAIR SIEBER: Thank you.

MR. KENNEDY: Okay. So our findings regarding the facility siting criteria was that this reactor is still appropriately sited. This, you know, agrees with the previous conclusions and that the hazards related to this site are not expected to pose a significant threat to the safe operation of the facility during the period of the renewed license.

CHAIR SIEBER: To what extent did you examine the traffic on 270 to determine whether there was a hazard based on cargo-carrying vehicles? It seems like every time I drive there, there's some kind of an accident someplace. Was an analysis done by anybody that refers to recent traffic patterns? When I first come down here, there was not 13 lanes across. It was only four.

MR. KENNEDY: I believe there was some. We did look at the types of traffic and some of the types of accidents, and there hadn't been -- based on the history, there hadn't been any severe accidents that we felt could have any impact on the reactor building itself.

CHAIR SIEBER: Okay. Actually, the reactor itself is set back from the highway a pretty good distance, and other than high explosives, gasoline trucks, or something like that, probably wouldn't have any impact on the site at all.

MR. KENNEDY: That's correct. The 270 is entirely outside their emergency planning zone. That is at least 400 meters.

Okay, for our review of the structure systems and components, we included the reactor confinement building, fuel design, core support structures, reactivity control devices, and really structure systems and components that we felt were most important to safety.

In terms of the engineered safety features, we did look at the reactor building ventilation, and the confinement building itself is somewhat of an engineered safety feature. Auxiliary systems, again, included the fuel pool and fuel handling and fuel storage.

So based on our review, we did find that the design bases that were originally used to design and construct the structure's systems and components remain valid for the reactor as it operates today. Yes?

MEMBER STETKAR: And, again, I recognize

that the criteria for the research reactor is much different than power reactors, but was any attempt made by either your staff or the licensee to perform things like failure modes and effects analyses to examine interdependencies among SSCs, among systems, and in particular with respect to support systems, AC power, DC power?

I noted that there are a lot of pneumatically operated valves, so compressed air and things like that. Has anything like that been done in terms of reexamining those types of inter-system issues in light of what we understand today compared to what was done in the original licensing area?

I'm not talking about a formal probabilistic risk assessment. It's more of a, you know, if you want to characterize it as a deterministic failure modes and effects analysis. If this fails, what are the consequences?

MR. KENNEDY: Well, I did not do a interdependency analysis any more than to the extent that looking at the analyzed accidents in terms of loss of electrical power, how do valves fail. You know, if they lose power, do they isolate? Do they remain open?

MEMBER STETKAR: But you did look at that?

MR. KENNEDY: I did look at that, yes.

MEMBER STETKAR: Okay. Good.

MR. KENNEDY: Yes. So from that, I guess, from that respect, I did do that kind of analysis and looked at what the licensee had to say about it in their application. I felt that the accidents that were analyzed, they did encompass the full range of failures in that respect.

MEMBER STETKAR: Okay. Well, thanks.

MEMBER BROWN: I have -- just to segue back to one other question on the single channel, the idea that you all's accident analysis on the -- what do you call it, the MHA?

MR. KENNEDY: Yes.

MEMBER BROWN: Whether it's credible or not, it was the accident you analyzed, and if that's an initiating event, that comment you made earlier, and then you assume everything works after that. That means your confinement, I guess, the ventilation shuts off, and nothing gets out of the -- theoretically gets out of the building except residual stuff.

MR. KENNEDY: That's --

MEMBER BROWN: Is that correct?

MR. KENNEDY: That's somewhat correct. The idea is to recirculate the building air to filter it and to provide a controlled release pathway out of the building, not to try to actually contain everything. It's to have a --

MEMBER BROWN: If that system had failed and you didn't have a controlled release, was there any analysis done of that, that now you had uncontrolled release as a result of those systems not doing what you said it was supposed to do, since it's only one system?

MR. KENNEDY: Well, I think I'd like to defer this question to Al Adams. I think he can better answer this question.

MR. ADAMS: Hi. Al Adams. I'm a project manager in the research and test reactor group. Post-911 we performed security analyses, security assessments of the facilities, including NIST, and those assessments, I think, address some of the issues you're talking about.

A question was asked earlier about accident analyses or accident scenarios that were intentional, and the answer is that's an example of intentional analysis. We assume the adversaries entered the facility and performed malicious acts.

In the analysis that we did, we assumed a greater amount of core damage than you see in the MHA, and we assumed that engineered safety features did not perform their tasks. We looked at the doses from that event at the site boundary, and they were within 10 CFR Part 100.

MEMBER BROWN: What, roughly, you know, like before you had a number of 7 mrs what was --

MR. ADAMS: These analyses were safeguards information, and that's about as far as I feel comfortable going in a public forum. I can tell you that the doses were less than 10 CFR Part 100 at the site boundary.

MEMBER BROWN: Okay, so that's -- is that kind of an answer to the -- okay, so that's part answer. I mean, I was thinking about that after I went through the stuff, also, the single-channel aspect, you know, no failures.

MEMBER BLEY: I have some other questions I want to get to in a bit, but that may preclude their importance.

MR. KENNEDY: Thanks, Al. Also, our findings were that we believe that the structure's systems and components can be expected to continue to provide for safe reactor shutdown and operation.

In terms of reactor characteristics, we looked at reactor control, neutronics characteristics, void coefficient, temperature coefficient of the moderator. We looked at a range of transient behavior and also checking fuel temperature to make sure that it remains well below the safety limit, and we looked at the margins by which the licensee has shown that their fuel will remain below the safety limit fuel temperature.

We found that, based on the analysis provided and our review, the neutronic and thermal hydraulic behavior provide reasonable assurance that this reactor can be reliably operated. They have the right types of control devices and good limits on the rate of reactivity, in addition, and sufficient shutdown margin and core access reactivity to reliably operate the reactor and also that the safety margins are adequate to protect the safety limit under all conditions.

That includes running the reactor well outside of its normal regime of operation at a -- running the reactor with both the -- all the temperature flow and the power level all at their limiting safety system settings still provide more than a margin factor of two below reaching departure from nuclear boiling, critical heat flux ratio, or the onset of flow instability.

MEMBER ABDEL-KHALIK: Now, the licensee presented earlier that the mpc is about -25 pcm per degree, and the moderator void coefficient is a -30 pcm per liter. Are these numbers consistent?

MR. KENNEDY: Consistent with?

MEMBER ABDEL-KHALIK: Are the two numbers consistent?

MR. KENNEDY: Those are the numbers that are consistently used throughout the analysis.

MEMBER ABDEL-KHALIK: But are they internally consistent?

MR. KENNEDY: I guess I can't really answer your question. All I can say is that they generally would take the -- as I mentioned, they would search the core for the point where the moderator void coefficient is smallest or most conservative and likewise for the temperature coefficient.

MEMBER ABDEL-KHALIK: But what is the core average void coefficient?

MR. KENNEDY: I don't have that number with me.

CHAIR SIEBER: You mean during normal operation?

MEMBER ABDEL-KHALIK: Right.

MR. KENNEDY: I think the point is that

their analysis is always using these worst case coefficients, even though they would -- they're more conservative than what the averages would be.

MEMBER ABDEL-KHALIK: But they have to be internally consistent, regardless of which direction of conservatism they're using, and the question is whether -25 pcm per degree C and -30 pcm per liter are consistent.

MR. KENNEDY: I will have to get back to you on that.

MEMBER ABDEL-KHALIK: Okay.

MR. KENNEDY: Unless our technical contractors have anything that they -- if Jim Wallace has anything that he'd like to add about that. Okay.

We also looked at electrical power systems. This includes normal -- power for normal operation, so they did have redundant feeds into the building. Also, for the emergency power systems they do have several sources of emergency power or backup power. They have both batteries and two diesel generators.

We looked at the loads that needed to be supplied during the loss of offsite power, made sure that those were appropriately specified in the technical specifications, and we looked at how long they would need emergency power and how long they would have it for, and we found that they do have adequate systems for electrical power in place to maintain a safe shutdown and also to operate the facility under normal conditions.

MEMBER STETKAR: I've just got a quick one in here. You said earlier that you looked at failure positions of valves on loss of power and things like that, and I know they have analyses for the loss of offsite power, initiating that. Did you look at loss of DC power in particular?

Because of the fact that they only have the one DC power supply that supplies everything, instrumentation, control, et cetera, and there is not a safety analysis, an accident analysis for that type of event, I was curious whether you thought about that in your review or whether you looked at it in terms of these dependencies.

MR. KENNEDY: Yes.

MEMBER STETKAR: Oh.

MR. KENNEDY: We did. We did look at it, and we did a review of the licensee's analysis, as they mentioned, of the loss of their shutdown cooling pumps and the loss of all power and found that the cooling by natural circulation was adequate in that case in that there aren't any systems that would be required.

MEMBER STETKAR: And when you say cooling by natural circulation, you mean cooling by natural circulation just within the vessel itself --

MR. KENNEDY: Yes.

MEMBER STETKAR: -- or the biological shield, not accounting for any of the secondary systems?

MR. KENNEDY: That's correct.

MEMBER STETKAR: Is that correct? That's correct?

MR. KENNEDY: Yes.

MEMBER STETKAR: Okay. Thank you. By the way, does that natural circulation cooling require availability of the biological shield cooling system, because that comes off, I think, part of the primary coolant purification system? I might be not remembering that correctly, but --

MR. KENNEDY: No.

MEMBER STETKAR: No, so it's just completely passive.

MR. KENNEDY: Yes.

CHAIR SIEBER: It's just a heat sink, yes.

MEMBER STETKAR: Okay. Thanks.

MEMBER ABDEL-KHALIK: So how hot would the vessel have to be in order to actually remove decay heat by natural circulation?

MR. KENNEDY: How hot would the coolant actually --

MEMBER ABDEL-KHALIK: Would the vessel itself have to be? I mean, after all, you sort of either convectively cool to the outside or radiatively cool to the outside.

MR. KENNEDY: That is a temperature. I don't believe I have that temperature.

MEMBER STETKAR: That's a good question, because it gets into if it's still communicating with the primary relief valve, which it would be.

CHAIR SIEBER: You don't remove heat unless there's temperature difference.

MEMBER STETKAR: That's the question. You know, the pressure and temperature inside the vessel, if the pressure is above the relief valve set point, you might not get the circulation set.

CHAIR SIEBER: Well, you've got boiling if you've got that.

MEMBER STETKAR: Then you require make-up, which gets back into the --

MR. KENNEDY: Okay. We also looked at

their experiment program. We did look at the experimental facilities, also accidents involving experiments, especially reactivity, reactivity addition accidents, and reviewed to make sure that failures of experimental facilities wouldn't adversely impact the reactor core itself.

MEMBER RYAN: It might be helpful for the full committee meeting if you could give a couple of examples of experiments that have been reviewed of recent vintage, and, you know, I think -- I mean, because you think of an experimental reactor.

There's a wide range of folks doing a wide range of things, and I think the committee would benefit to have a couple of examples of that process. We did this, you know, kind of an experiment with this much excess reactivity or this kind of an exposure to targets, whatever it might be, so you get a better feel kind of for all the aspects, the reactor part as well as the radiation protection part and the ALARA program and how all that worked. I think -- you know, I sure accept your judgment in the findings, but it would be nice to have a little bit more detail to exemplify that.

MR. KENNEDY: We didn't specifically look at each -- at recent individual experiments. We looked at the limits on experiments that are imposed by the technical specifications, and then it's up to the licensee to, through their internal review process, to make sure that their experiments that they would like to change -- they would have to conduct a 50.59 review and --

MEMBER RYAN: Well, we need to hear about that, whoever is doing it, because without understanding that, it's very hard to accept the findings. It sure is to me.

MR. KENNEDY: Okay. I mean, if the licensee determined that they needed an amendment to their license, then the NRC would review that experiment. Otherwise, we would review their change documentation.

MEMBER RYAN: And I think -- I appreciate that. That would be a different license amendment, but under the current relicensing they're doing that. I don't know how many folks here have actually, you know, understand how the licensee does its process at that level of detail without a little bit more discussion of that, so if we could hear from somebody, whether it's the licensee or the staff, in your review of their work, it would -- I think it would be very helpful to hear more about it.

MR. KENNEDY: Okay.

CHAIR SIEBER: Yes, I think, just to expand a little bit, the experiments should not introduce additional special nuclear material that would contribute to the reaction. I think they're -- and I think that's the case as far as the requirements are concerned. In addition to that, there is some discussion in the literature about some experience having a aggressive chemical reaction.

MEMBER RYAN: And there's probably others that, you know, address contamination control and counter measures during experimental handling and all that kind of stuff.

CHAIR SIEBER: Well, that's -- the health physics issues, I think, are significant. I think the introduction of materials to the beams represents, if anything, a additional neutron absorber.

MEMBER RYAN: Sure.

CHAIR SIEBER: So from a reactivity standpoint, I would not expect any increase in reactivity.

MEMBER RYAN: But I'm just trying to get a sense of what the licensee's process, I guess, is to do that. I understand that's enveloped by the conclusion that the staff is offering.

MEMBER SHACK: Well, I think in response to a question that Said asked earlier from the licensee, though, there was some hint that some of the experiments involve in-core. I mean, you know, the neutron beams are one set of experiments, but, you know, if there are core experiments, I think, you know, we need to be a little bit perhaps more careful about just what those involved, and so that's the particular experiments, at least from the reactor safety, I think we would need to focus on.

CHAIR SIEBER: Right.

MR. KENNEDY: Their technical specifications do have requirements for the types of materials that can be introduced into the experimental facilities, and those deal with some of the issues that you've mentioned, and so the staff also looked at the administrative controls for experiment review and approval. That's what we've been discussing, that they do put experiments through the proper safety review committees prior to introducing any new experiments to the facility and also, as part of the inspection program, that they are doing their reviews properly and documenting them.

And so we found that the experiments should not pose a significant risk to safe operation of the reactor, and this includes, again, reactivity additions or any types of mechanical reactions or mechanical impacts that could damage the reactor core support structure or the core itself. And we also found that new experiments will be properly reviewed and approved before being implemented at the facility.

MEMBER BROWN: Have there been any accidents? Maybe I missed the question or an answer. Have there been any accidents involving experiments?

MR. KENNEDY: Not that I know of.

MEMBER BROWN: Or you listed it but didn't get to it. Okay, so there have been -- of all the guys that have come in, guys, people, for the last 40 years, there have been no accidents. Nobody has messed up, or they've had no --

MR. KENNEDY: Not that I know of.

MEMBER BROWN: Dumb question. I'm just --

CHAIR SIEBER: I don't remember reading that anyplace.

MR. KENNEDY: The reason we do --

MEMBER BROWN: No over-exposures, nothing like that in terms of conducting an experiment?

MR. KENNEDY: Not that I'm aware of.

MEMBER SHACK: I think the licensee, perhaps, can respond more forcefully to that.

MR. RICHARDS: There are no accidents involving the experimental programs. We haven't had any over-exposures, either, that I -- so. The review process and the handling of everything have been pretty rigorous. Wade Richards, NIST.

MEMBER BROWN: I -- somebody looks like they want to say something over there behind Sam.

MEMBER MAYNARD: I'm sure you'd have to be a little careful on how you define accident. I mean, I'm sure not everything went perfect, so I think part of this would have to come down to what you classify or call an accident versus a variation --

MEMBER RYAN: A variation from experimental design. I mean, that's a fair comment, I think.

CHAIR SIEBER: Unexpected.

MR. KENNEDY: Well, for the purpose of this review, we looked at accidents as they're presented in the accident analyses, which was the worst case possible failures of accidents in terms of reactivity addition to the reactor.

MEMBER BROWN: Based on the experiment, not based on a reactor control event but based on an experiment introduced.

MR. KENNEDY: Correct, both reactivity -- there are limits on the reactivity of individual experiments, both for in-core and also for the pneumatic transfer system, so we looked to make sure that the pneumatic transfer system wasn't going to be able to introduce reactivity any faster than what was analyzed, as well, and we don't -- again, we don't postulate who moves the experiment or what happens to it. We just look at what is its reactivity, maximum reactivity, and then it moves.

MEMBER MAYNARD: I think the applicant might want to add something here or say something.

MR. ROWE: Going back on your -- excuse me, Mike Rowe from NIST. Back on the previous question, if you're asking has their been an accident in the sense of anything in-core or something failing, the answer is no, but you didn't include in there a question of over-exposures, and we had one over-exposure. It was -- did not exceed regulatory limits, but it exceeded administrative limits.

We did a dose reconstruction, a careful dose reconstruction and investigation to understand why, but that, I believe, is the only one that I remember, and my colleagues are agreeing with me, and that goes back 35 years, so pretty close to the beginning, but I won't guarantee the first five years.

MEMBER BROWN: Okay.

MR. KENNEDY: Okay. We also reviewed radiation protection, and we looked at the sources of radiation, what kinds of effluence the facility was generating, the wastes that they are generating and would be generating, the personnel monitoring program, also their environmental monitoring, which includes the sampling of vegetation and soil and water, and we looked at the administrative controls they have for radiation protection.

MEMBER RYAN: A couple of questions here.

MR. KENNEDY: Yes.

MEMBER RYAN: You know, with the unavailability of B and C disposal for low waste Class B and C, is there any accumulation of waste --

MR. KENNEDY: Well, for --

MEMBER RYAN: -- that doesn't have a home, and do you generate any mixed waste that doesn't have a home?

MR. KENNEDY: In terms of the license renewal review, we looked that they have the appropriate stipulations in their license that they need to properly dispose of that waste.

MEMBER RYAN: There's no access to B and C.

MR. KENNEDY: We don't look to see how they're -- where they're going to send it. We just look to see that they are required to dispose of it properly.

MEMBER RYAN: But that assumes that there is a place to dispose of it. If they have nowhere to go, then they're going to accumulate it, so it seems that you want to look at their ability to accumulate it in a controlled way.

MR. KENNEDY: They do --

MEMBER RYAN: I would assume they accumulate some just from the mechanics of shipping waste, but if there is no B and C disposal in the United States, which at this point there isn't, there's nowhere to go.

MR. KENNEDY: They do have limits on the amount of material they can possess, and they do have facilities to store that material, and if they were to reach those limits, then, you know, from an operational standpoint they have to stop.

MEMBER RYAN: Okay. Again, a little bit more detail on, you know, what the profile of all that is from a licensee's perspective would be helpful, I think, for the full committee to get some understanding of that.

MEMBER MAYNARD: I'm not sure that's correct, though. Usually, you have limits, and you have time frames, but at the end of that time frame, if there is nothing available, then it can be relicensed or reevaluated for additional storage.

MEMBER RYAN: Well, the licensee always has the option in applying for more storage capacity, but --

MEMBER MAYNARD: And I'm also not sure that they are restricted to the commercially available waste disposals. I'm not sure. They may have --

MEMBER RYAN: I didn't restrict myself to that.

CHAIR SIEBER: The yard, I think, stated that low-level waste was provided for by a commercial disposal site using, you know, regular dock transportation, so in that respect it's like a power reactor. Spent fuel, however, since there's HEU, has to be taken care of. You can't accumulate a lot of it, and it's sent to -- the federal government retains ownership of the fuel throughout the process.

MEMBER RYAN: Sure. That's not dissimilar to a commercial power plant, but there is no option in the United States for Class B and C disposal --

CHAIR SIEBER: That's true.

MEMBER RYAN: -- except for the states in the Atlantic Compact.

CHAIR SIEBER: Yes, mixed waste.

MEMBER RYAN: So they have B and C waste on site, they can do nothing with it, so I'm just asking the question is that accumulation of waste, you know, an issue or a non-issue. I'm guessing it's probably a long-range issue, if anything, but it would be nice to hear a clarification.

CHAIR SIEBER: Well, the question really is do you have any B and C waste?

MR. BROWN: Dave Brown, Health Physics. We do generate Class B and C waste, and we do have plans for provisions for storing through the life of the facility onsite if we do not get an option for disposal.

MEMBER RYAN: And does that challenge your off-site dose or bounding dose calculations in any way? Based on the size of the site, I'd guess no.

MR. BROWN: No, it doesn't.

MEMBER RYAN: Okay. Thanks. That's helpful.

CHAIR SIEBER: Thank you.

MR. KENNEDY: Okay. As far as findings regarding radiation protection, they do have a sufficient radiation protection program to keep doses below the regulatory limits and ALARA. They have appropriate controls in place to prevent uncontrolled releases or release of material in excess of regulatory limits, and there is reasonable assurance that they will properly handle and disposition the radioactive waste.

MEMBER RYAN: Just one more word on tritium, if I may. I just point out that power reactors, several around the country, have identified tritium at below the regulatory limit. It still is a big issue, and I guess I haven't heard yet enough information about the old wells, and hearing that they were domestic wells troubles me a little bit, because domestic wells typically are shallower than you're interested in in groundwater wells.

So, again, I'd ask for a little bit more detail on what that is all about and, one, what the licensee has evaluated in that area to end up at the program you're operating for that environmental monitoring and why you discounted detailed groundwater analysis, because it sounds like that's what you've done, and then how that's factored into the staff's review.

MEMBER BLEY: Before you get into the accident analysis, I'd like to ask you a couple questions, because the things Harold brought up earlier were a little troubling to me, and your responses and those of the licensee raise some questions, but I've gone back and reviewed again the SRP while you were talking.

This idea that, in the accident analysis, that's the one failure and you don't have another single failure seems to me it's not quite the way -- it's not quite the way I read the SRP, so let me tell you what I read, and then if you'd comment on it, I'd appreciate it.

Looking through it, there is this single failure criterion in the seismic criterion and some others for both the RPS and radiation protection, and those both say that you have to be able to take a single failure on either of those systems and provide the function, including in response to an accident that's analyzed in your accident analysis.

For ESFs, the wording is not the same, but there is wording that almost gets one to the same place, I think, which says you have -- they have to work on loss of electric power. They have to be operable -- and maybe what operable means is the issue there -- for any accident that occurs, and you have to have reasonable assurance of reliable operations if they're required. Reasonable assurance to me is either something like single failure or some kind of reliability calculation.

In the SAR, it points out that the ESF, especially the emergency cooling system, does have redundant valves, so they can survive a single failure, as well, but they have manual operation required for them.

All this together tell me that you have the equivalent of being able to take another single failure, in fact, multiple failures, one in each of those systems, after your accident. I wonder if you think that's true.

The last thing is the one part that -- where what I've heard doesn't seem to align with what I see in the SRP is you have to be able to survive seismic events, and there are other statements that say you have to be able to survive failures caused by other systems, which kind of says to me that those masonry walls that -- unreinforced masonry walls, if they could take out a secondary system that would affect the first system, you're not meeting what you're supposed to be meeting in the SRP.

So it's in two pieces. The first piece is, in general, you really, I think, through the SRP are covered for the failure after the event itself, and the second is is your seismic -- is their seismic analysis up to what's required?

MR. KENNEDY: The first question, yes, we think we are covered by single failure criterion.

MEMBER BROWN: The SRP requires a single failure criterion.

MEMBER BLEY: The SRP requires that both RPS and rad protection will continue to operate given a single failure within them, and that's equivalent to taking a second failure after the initiating event, it seems to me.

MR. KENNEDY: Failures of the reactor protection system, failure of a channel, would lead to scram on the reactor, so the reactor would shut down if any channel goes out.

MEMBER BLEY: On that kind of thing it does.

MR. KENNEDY: And then they have redundant channels.

MEMBER BLEY: It's the redundancy requirement that I think covers you, so --

MR. KENNEDY: Yes, they do --

MEMBER BLEY: You don't -- it doesn't seem that we'll answer this on the spot, but maybe you can think about that one a little bit, because the answer that we don't have to take a single failure following the initiating event, because it's not required in the SRP I think is an over-general -- I don't think it's right.

MEMBER RAY: Dennis, during the break -- let me speak here. During the break, I was approached by one of the licensee, and the thing that triggered my interest was there's a statement in the SER that reads, "The normal air monitor channel, irradiated air monitor channel, and stacked monitor channel," each of those words in the singular, "control relays in the major scram circuit," and so on and isolate the confinement building.

I was told that those are actually, even though each one has a different name, are redundant of each other and required to be operable, at least two out of the three. So even though it's described the way it is, making it sound as if there are single detectors in these different places, they have the effect of being redundant. Now, that's what I understand to be the case at the moment, and I'll just leave it there.

CHAIR SIEBER: Well, if that's not the case, somebody should tell us.

MR. RICHARDS: That is the case.

CHAIR SIEBER: Okay. Thank you.

MEMBER BLEY: The second part of my question was there seem to be requirements for seismic that it's not clear to me you're meeting. If people -- if we don't even know where unreinforced walls are, how do we know we can -- the design can survive the seismic events that could happen?

MR. KENNEDY: Well, in this area we did rely a lot on the past work that had been done in licensing.

MEMBER BLEY: I don't know what that means.

MR. KENNEDY: That means that we looked back and made sure that the assumptions that were made when the original seismic analysis was done and the analysis of the last renewal, that those assumptions still held true.

MEMBER BLEY: There is -- you know, I know this is a research reactor and it's not a power reactor, but I think there can be learning from what happened for power reactors, and one of the things that was learned in the mid-seventies, early eighties was in the power plants there were a lot of unreinforced concrete walls that hadn't been analyzed, because nobody really thought about them. They were just partitions, and when people looked at them, they found those could either take out safety systems, or they could take out systems that support the safety systems, cascading, leading to a failure.

Seems to me by your SRP you need to be able to survive those kind of events, and I don't think anybody looked at them back in the sixties, so I'm not sure that taking what was done then takes care of it now, notwithstanding what Al said might cover you in all cases, but as far as what you should have been reviewing for this, I'm not convinced that we got it from what I've heard today. I'd say let's go on, but I just wanted to --

MR. KENNEDY: Okay. I will keep that in mind.

MEMBER STETKAR: Related to seismic, is there a specific safe shutdown earthquake or design basis earthquake, however you want to characterize that, that the design is required to meet? I saw .1 g kind of bandied about in the Safety Analysis Report. Is that a formal design basis earthquake requirement for this facility, or is that simply a piece of information?

MR. KENNEDY: That was what the facility was -- that was one of the licensing criterion. We looked to make sure that the hazard in the area was well below that.

MEMBER STETKAR: Okay, but that's a frequency of hazard. It's not survivability of the equipment.

MEMBER SHACK: But even the frequency, you know, somewhere it says it's a .08 g is two percent in 50 years.

MEMBER STETKAR: I have those numbers, if you look up the USGS stuff, and that is indeed supported by the current U.S. --

MEMBER SHACK: But, I mean, still it's not the kind of safe shutdown earthquake we would use for a power reactor.

MEMBER STETKAR: No, no. You could not apply the frequencies here in the power reactor.

MEMBER SHACK: Right, I mean, and we're not even close.

MEMBER STETKAR: No.

MEMBER SHACK: So, you know, it is kind of curious just what the basis for the seismic requirement, whatever it is, is.

MEMBER STETKAR: Because you're right. The frequencies, you know, could not be --

MEMBER SHACK: We would die laughing.

MEMBER STETKAR: Yes. That's correct, but given the fact that there is some rationale that a .1 g is acceptable, then the question is, indeed, will the equipment inside the facility, including those masonry walls, if there are any, survive under a .1 g, things like battery racks and, you know, all that kind of stuff.

CHAIR SIEBER: Well, two over one.

MEMBER STETKAR: Well, and two over one that Dennis was talking about.

MR. KENNEDY: Okay.

CHAIR SIEBER: Well, if the staff can't give us an answer right now or the applicant, I think that that would be something we'd like to learn about when we have our final meeting, so if somebody could prepare a position on that, I'd appreciate it.

MR. KENNEDY: I would have to look into where additional walls have been put up. You know, the reactor as it was designed was designed to safety shut down in that case, a seismic event, so it's a matter of whether these -- I'm not sure what kind of --

MEMBER SHACK: You know, frequently what happens is you design the safety system to take the .1 g, and you sort of forget that the safety system can be disabled by something else, so, you know --

MEMBER BLEY: And not just walls. Other things can fall down.

MEMBER SHACK: Other things can fall down, right.

MEMBER STETKAR: Other electrical cables and things like that. For example, whatever credit is taken for ventilation systems, if they can't function because failure of a block wall has torn apart electrical cabling or something like that --

CHAIR SIEBER: Or duct work.

MEMBER STETKAR: -- or duct work.

MEMBER SHACK: And what's confusing here, you know, in power reactors we have a very specific scope for what's involved in license renewal. In this case, it seems more like it's more fair game to look at everything in the licensing basis.

MEMBER STETKAR: It's a re-licensing.

MEMBER SHACK: It's a re-licensing, yes. You know, we're not just looking at the passive long-lived components and whether they're aging, in which case the answer is, you know, not much, but, you know, we are re-licensing, and --

MEMBER STETKAR: Given what we know today.

MEMBER SHACK: Yes, the seismic basis seems very strange, but Al's answer is helpful.

MEMBER BROWN: It almost -- yes, the answer almost makes it sound like it doesn't make any difference based on that one -- I find that hard -- I don't want to say it's hard to believe, but, I mean, I'd like to be able to demonstrate why that's the case. I want to make one other observation, if you're finished, John. Go ahead.

MEMBER STETKAR: No, go on.

MEMBER BROWN: You made the comment about if you had a lost of an instrument, protection instrument or the safeguards, ventilation, whatever the radiation monitors are, that it scrams the reactor, but I don't -- a loss of an instrument can be defined in many, many ways, and instruments can fail such that they are measuring and putting out a normal output.

So when you're talking, if you're just saying if they lose power they will, you know, they will scram, that's one thing, but these are electronics. They can fail such that you get a normal output, and they will not scram anything. They'll just sit there, and you're thinking you're happy as a pig in a mud wallow, and nothing happens.

So just be -- I just -- it's just an observation. You've got to be careful with that statement about loss of instrument scrams the reactor, because it will not in most cases, as a matter of fact.

CHAIR SIEBER: Why don't we move on?

MR. KENNEDY: Okay. Yes. So accident analyses, we looked at a maximum hypothetical accident, again, no assumed initiating events, just you all of a sudden have availability of fission product material in the reactor coolant. We found again that the doses at the site boundary resulting from this accident were only a fraction of regulatory guidelines in Part 100 and also below the limits in Part 20 for members of the public and for facility personnel.

We looked at reactivity insertion, both the startup accident with a continuos control rod withdrawal, as well as the reactivity insertion due to almost a step insertion of the maximum reactivity allowed for an experiment.

We looked at loss of coolant, and this is a complete loss of all the reactor coolant, assumed to be some sort of large pipe break that all the coolant drains out of the reactor and is collected in the reactor sump, and we also found this accident not to challenge fuel integrity, which I think in some ways can cover other accidents that could happen such as the reactor vessel cracking, leaking all the coolant that way, or a seismic event causing loss of the primary coolant system.

We also looked at several different types of the loss of coolant flow, a pump seizure and a throttling of coolant flow to the inner and outer plenums and loss of the shutdown cooling pumps, and the licensee also described their analysis of the misleading of fuel, and we have reviewed that, as well. Yes?

MEMBER STETKAR: I should know this, and I forgot. Did the licensee perform any additional accident analyses in response to any of your RAIs, or did you simply review the existing, you know, cadre of accident analyses? In other words, did your review generate additional?

MR. KENNEDY: I don't remember that we asked for any --

MEMBER STETKAR: I couldn't remember.

MR. KENNEDY: -- additional accident analyses. We did ask for clarification on the analyses.

MEMBER STETKAR: Yes, no, I know that. I was -- okay, thanks.

MR. KENNEDY: Yes.

MEMBER STETKAR: Thanks.

MR. KENNEDY: The types of accidents they analyzed were consistent with what was in our review plan, and we didn't go outside the bounds of what was in the review plan.

MEMBER STETKAR: That's fine. I just couldn't -- thank you.

CHAIR SIEBER: I don't want to leave the impression that the licensee has not been active in updating analysis and performing additional analysis, because it's my understanding, and I can -- you can correct me if I'm wrong. It's my understanding that this work has been done and considered appropriate, for example, the updating of codes, and, for the record, everyone is nodding their head yes.

MR. KENNEDY: I have already discussed these findings below, doses below Part 100 guidelines and the -- well, the limiting safety system settings, which are the initial values that they use in their accident analyses do provide adequate safety monitoring to protect the safety limit on the fuel.

MEMBER STETKAR: On your LOCA analysis, the LOCAs do, however, require that the emergency sump pump must eventually kick in and recirculate the D_2O, right, after some period, after the --

CHAIR SIEBER: Two hours or something.

MEMBER STETKAR: Yes, two and one-half hours or after the tank screen. Is that correct?

MR. KENNEDY: There is also, I believe, a quick-connect where they can hook up to city water and just pump normal water through the core if needed.

MEMBER STETKAR: That's not a safety system, though.

CHAIR SIEBER: That's correct. That's a backup to a -- a non-safety back up to a --

MEMBER STETKAR: But your LOCA analysis that you mentioned that they've done takes credit for that. There is one emergency sump pump.

MR. KENNEDY: Yes.

MEMBER STETKAR: And maybe this is a question to the licensee. Does the LOCA analysis -- well, this gets back to the single failure thing. You don't -- apparently, you don't have to assume a single failure of that sump pump. Okay.

MEMBER SHACK: But he does have his possibility of a connection.

MEMBER STETKAR: Yes, there is, but that's a manual backup, not a safety -- I'm trying to get my hands around the scope of the analysis. Thanks.

MR. KENNEDY: You're welcome.

CHAIR SIEBER: Okay.

MR. KENNEDY: Now, in our review of the technical specifications, we looked at these six factors here that are all required by 50.36 in the regulations. There was some change to how the licensee went about presenting their safety limit, and staff reviewed that and found that this was acceptable.

We did find that the tech specs as they stand now do meet the requirements of 50.36 and that they do provide reasonable assurance that the facility will and can be operated as analyzed in the Safety Analysis Report.

MEMBER STETKAR: Well, it may lead into the next topic, but I noticed there was quite a bit of discussion in the RAIs back and forth about the fact that the tech specs do not include any limits on primary system chemistry or fuel pool chemistry --

CHAIR SIEBER: Right.

MEMBER STETKAR: -- and that -- and apparently the staff is satisfied that whatever programs are in place are sufficient to assure the fact that long-term effects of chemistry are being monitored and controlled. I was curious about how you developed that sense of comfort, let's say, without any specified limits in the tech specs or any required surveillance to maintain those limits.

MR. KENNEDY: Well, in terms of the primary chemistry, they do have a closed system, and they do monitor both pool, fuel pool and primary chemistry as part of their program.

MEMBER STETKAR: That was mentioned in the SER, but I didn't -- there's no requirement in the tech specs to say how frequently that monitoring is done or any limits on --

MR. KENNEDY: That's correct.

MEMBER STETKAR: Okay.

MR. KENNEDY: The administrative controls in place for primary coolant seem to show that they were not going to have any big change in chemistry in a rapid manner and that they would -- they would be maintaining the purity of their D_2O. They do have purification systems, as well.

MEMBER STETKAR: But over in the power reactor side of the game, all of those statements are precisely true, also, and yet we have chemistry limits. I mean, you know, there are closed loop cooling systems, and there are administrative controls in place, and there's, you know, a purification system with demineralizes and --

CHAIR SIEBER: They also have a purification system.

MEMBER STETKAR: Yes, they do, but my point is on the power reactor side, despite all of those completely analogous systems, there are still chemistry, specific chemistry limits for priority --

CHAIR SIEBER: The environment --

MEMBER STETKAR: -- for the primary side and even in some cases the secondary side.

MR. KENNEDY: I'll look into that.

MEMBER STETKAR: It was one of these things, you know, again, thinking in terms of long-term effects of what's a cumulative effect of small changes in chemistry, not necessarily the catastrophic, one-time only requiring a huge cleanup that you know about. It's gradual changes over time that might be a concern and recognizing that their normal administrative controls should find that, but from a regulatory perspective, if there is no specific limit or a specific requirement in the tech specs, you know, you have less -- you as a regulator have less control over that situation.

MR. KENNEDY: Okay.

MEMBER STETKAR: Thanks.

MR. KENNEDY: Let's see. For the prior years of reactive components, we looked at fuel. We felt the fuel was in the core for such a short time that we didn't have a problem with prior use of the fuel.

We've talked a lot about embrittlement already. I don't know if you want me to say anything more about this other than I looked at the licensee's analysis. I looked at the reference documents. I looked at the fast flux to thermal flux ratio, and I felt that their analysis showed that they weren't going to approach the limit, and even if their vessel were to break, it's analyzed as a loss of coolant accident already.

MEMBER ARMIJO: Do they have a surveillance program, a vessel material surveillance that they periodically take out just to verify that it's --

CHAIR SIEBER: Visual and thickness.

MEMBER ARMIJO: What?

CHAIR SIEBER: Visual and thickness.

MEMBER ARMIJO: I'm just talking about mechanical properties to verify that the embrittlement is --

CHAIR SIEBER: No.

MEMBER ARMIJO: Or is changing as expected.

CHAIR SIEBER: No.

MR. KENNEDY: No.

MEMBER ARMIJO: It's just based on --

MR. KENNEDY: Just based on calculation.

MEMBER ARMIJO: Experiments done before under --

CHAIR SIEBER: Right.

MEMBER ARMIJO: And it's assumed that this material is behaving exactly as expected, but it's not verified.

MR. KENNEDY: That's correct. It's not verified.

CHAIR SIEBER: Okay.

MR. KENNEDY: Also, the licensee mentioned that they do switch out their control rods, so we didn't see any real aging issues there, and we felt that the surveillance requirements in the tech specs would help to -- would provide reasonable assurance that they would catch any types of, you know, slow drifts in their equipment in terms of its performance. And, with that, I'm going to quickly turn it over to Johnny Eads for a discussion of staff and inspection history.

MR. EADS: Seeing how we only have nine minutes, I'll be very brief. Again, I'm Johnny Eads.

I'm the Branch Chief for Research and Test Reactors Branch B. My branch is responsible for the inspection and enforcement activities related to NIST and the other non-power reactors, as well as doing operator license examinations and issuances.

Quickly from the slides, our inspections are conducted at NIST twice a year per the inspection manual, chapter 2545, which is the RTR Inspection Program. Let me tell you what that is not. It is not a reactor oversight program. It's not the ROP. There are no performance indicators. There are no red, white, green findings. It's basically the old style of inspection.

CHAIR SIEBER: But you still have findings, right?

MR. EADS: We absolutely have -- we can have violations and non-conformances, follow-up items, unresolved items, the full gamut as you would see. Just -- we do not color code. We have no PRA as basis.

Since Part 54 does not apply to us or research and test records in general, we do not do scoping and screening inspections. We do not go out and do aging management program inspections.

We do standard inspections throughout the life of the facility, and the areas we cover include reactor operations and maintenance, the radiation protection program, environmental and effluent monitoring, reactor surveillance, the review and audit program, design changes, emergency planning security. It goes on. Fuel movement, experiments, all those areas are covered in our routine inspection program.

MEMBER STETKAR: In the SER, the version of the SER we had, it said that the emergency plan review was not completed at that time. I've forgotten the date of the SER, and it was going to be completed by mid-January of this year. Has that review been completed?

MR. KENNEDY: I'll answer that. Yes, it has.

MEMBER STETKAR: Okay. And also the operator training was in the same -- I think you mentioned that's an open -- that still remains an open item. Okay.

CHAIR SIEBER: Open item.

MEMBER STETKAR: Thanks.

MR. EADS: Yes, we do -- and when we do the inspection, we use the emergency plan as our base document and review against that document.

The next slide just talks about the

results. The slide says in the last two years the reactor inspection program has not identified any violations or non-conformances at the NIST facility. I can tell you I just quickly this morning pulled back five years' worth, and even going back five years we have not had any violations or non-conformances identified at that facility, and it has been a topic of conversation.

The fear from the facility is that since our inspections are not identifying violations, perhaps complacency may set in, and so the facility is aware of those issues and takes action, and so we, too, share that concern, but I would congratulate them on the lack of violations in certainly the last five years.

MEMBER BROWN: Have you all had any findings? I had a boss one time that said people are inspecting stuff, you know, periodically, whatever it is, and they never found anything that --

CHAIR SIEBER: Something wrong.

MEMBER BROWN: -- there's something wrong, and they ought to be fired, so I'm saying that with a little bit of jest, but your point was well taken about not finding anything.

MR. EADS: I could say either --

MEMBER BROWN: Are there any findings or any issues of which --

MR. EADS: As you read our inspection reports, and we provide you the last two years, you will see that we had items where we questioned what was done. We challenged the staff to provide documentation to support what they had done, and we satisfied ourselves that no violations had occurred.

The best example would be the diesel generator, emergency diesel generator starting batteries. If you'll look back, you'll find that inspection report where we went out, and we observed that they replace both diesel generator starting batteries during the same shift, and, of course, you should be where we were where the question is on operability where those emergency diesel generators continue to be operable, because we didn't see any documented testing until the next required interval some weeks or months --

MEMBER BROWN: They replaced the batteries and didn't test them?

MR. EADS: According to the paper we reviewed, and so that's the sort of findings that we're looking for. We're out there looking for those instances where equipment is taken out of service, where maintenance is performed, where activities that might affect operability are impacted.

So we did challenge them, and what you'll see in that inspection report is they supplied us basis that said that as part of their routine maintenance program for replacing those batteries, they do exactly that. They do test them. They just did not have the documentation of those individual tests.

And so I don't want you to think that because we haven't had findings or issues that we're not looking. Our reports describe areas that we examine. We see -- I'll call them out-of-normal. We don't write up a lot of them, but those that we believe are significant or should be taken seriously we do identify.

I did go back and looked at when was the last inspector follow-up item written, and unfortunately I had to go back to 2004 before we found an inspector follow-up item. Again, I believe that to be a reflection of the performance of this facility, and I can talk in a little more detail about that in this next paragraph, because based on that history we believe that the NIST facility has a very stable, well managed, safety-conscious operating program.

They've done -- had significant advances. I call out particular -- the engineering effort, it is significantly stronger now than it was in previous years. We see new designs change processes being implemented, new drawing controls being enhanced. They've added new shielding for ALARA installation.

MEMBER ABDEL-KHALIK: How are those new shield panels supported?

MR. EADS: They appear to be supported from the floor. If the facility would like to address that, I can tell you they appear to sit on the floor.

MR. RICHARDS: Are you referring to the ones we just put in?

MR. EADS: Yes.

MR. RICHARDS: Yes, there was a full engineering design done, and the shielding panels are all a bolted assembly with --

MEMBER ABDEL-KHALIK: My question pertains to the earlier question about the seismic response.

MR. RICHARDS: Oh, we looked at floor loadings in the Burns and Rowe calculations for floor loadings. We always do that before we put any loading on the floor.

MR. EADS: I think you'd have to see this reactor operating bay to understand why that would really only be the concern. There is no other equipment that stands in and around the shielding. It's really shielding for ALARA purposes that I don't believe you'd find threatened from a physical standpoint nearby, safety-related equipment.

That being said, this facility is not without challenges. I don't have a particular slide on those challenges, but I'll quickly mention one of the biggest. There are two aging issues. The first is the aging of their reactor operators.

When you look at their facility, they have 22 senior reactor operators. They have no reactor operators. They have 22 senior reactor operators. Of that, half of them have been at that facility operating that facility for over 25 years, so they have an extremely stable staff, and when you go beyond that, you get the far majority being at least 20 years of service at that facility.

Now, that is a challenge, because those people need to retire, and there needs to be a transfer of knowledge, and the facility has acknowledged that need for knowledge transfer and is working to address it. They have not perhaps moved at the pace we'd like to see in updating the procedures and procedure space, but right now with the experienced staff they have, they have no difficulty in following the procedures as written

One challenge that we haven't had a chance to talk about that you need to be aware of, and that's going to be in the area with the thermal shield. A piece of passive equipment, the thermal shield provides a thermal barrier to protect the biological shield.

Well, that thermal shield system has a cooling system, and that cooling system is leaking. They understand, or they are attempting to understand the mechanisms associated with that aging and are taking actions to address it.

We have confidence that they will continue to follow it, and we will continue to follow it as part of the inspection program, but I don't want to leave you with a sense that this facility is spic-and-span brand new. It is old, and it does have components that are aging, but the facility seems capable and willing to manage that aging.

CHAIR SIEBER: Okay. Thank you.

MR. KENNEDY: In terms of major issues, as I mentioned, there was a new version of the technical specifications that was submitted in response to REIs, and the staff spent a lot of time making sure that this new version still included all of the important conservatisms that were in the original technical specifications and that it meets the current regulations.

This change in large part was done in order to get the technical specifications in conformance with the newest guidance for the development of technical specifications, and there were updated analyses in the SAR that were also reflected in the technical specifications.

We did find some inconsistencies in the first version that we received. However, we did work with the licensee to make sure that we were able to remedy all of these inconsistencies in a way that ensured safety as well as allowing the appropriate operational -- allow the licensee to operate the facility as they intended to.

MEMBER STETKAR: Excuse me. I am not sure whether we have the absolute current version of the tech specs. There were a couple of different versions that came through the mill to us, so my questions might be answered because we're one revision behind.

I notice that the licensee took exception to several of the surveillance requirements in ANS-15.1. In particular, things that jumped out at me was that the standard requires quarterly testing of emergency pumps, and the licensee said, well, they only want to test, for example, the emergency sump pump once per year, and apparently the staff found that acceptable.

The rationale in response to an REI was that the pump has been checked for over 25 years without a failure, and therefore an annual frequency is sufficient. If you only test it once per year, and it's 25 years, that's only 25 successful starts without a failure. That wouldn't -- you wouldn't really expect to see a failure. If you saw a failure, that would be a really terrible pump.

So I was really curious about the staff's rationale for accepting the one-year frequency as opposed to the quarterly testing frequency, and as far as the shut-down cooling pumps, which are part of the safety systems, I guess, there's no -- I couldn't find any requirement for testing those in the technical specifications whatsoever.

And I had one other minor one, because in the SER the testing frequency for the diesels as quoted in the SER seems to be different than the testing frequency quoted in the technical specifications. Perhaps the licensee could help me on that one.

In the tech specs, at least the version we had, it said that each diesel shall be started quarterly. The SER cites a monthly testing interval. Do you test diesels once a month, or do you test the diesels once per quarter?

MR. MYERS: The existing technical specifications is once a month.

MEMBER STETKAR: You have to identify yourself.

MR. MYERS: Tom Myers, NIST. Under the existing technical specifications, it's once a month.

MEMBER STETKAR: Okay. That's existing, but for the new license technical specifications will be -- will they be tested once a month or once per quarter?

MR. MYERS: I think it's once a quarter.

MEMBER STETKAR: Okay. That's different than what's cited in the SER. The SER says tech specs, emergency power systems requires that each diesel generator be tested for automatic starting and operation at least monthly and under simulated complete loss of power at least annually. Should one diesel become inoperative, second diesel is started at least weekly.

The tech spec, the actual tech specs that I read, required a quarterly testing. If a diesel becomes inoperative, the operable diesel shall be started monthly and that the actual loss of load test be performed annually so that --

I was looking for consistency between the real tech specs and the testing surveillance requirements in the standard, and the pumps that I mentioned was one difference, and the diesels seemed to be another difference, and the numbers that I just heard seem to confirm the fact that that diesel testing will be under the new license performed quarterly rather than monthly.

MR. KENNEDY: That's correct.

CHAIR SIEBER: I think that that requires some additional justification to us, and I think it would be good for the staff to --

MEMBER STETKAR: Again, we can't resolve it now but as kind of a takeaway for the --

CHAIR SIEBER: Yes, but we now deviate from the standard, and we should not do that lightly.

MEMBER STETKAR: Well, and what's the rationale for that?

CHAIR SIEBER: Right, and I think that it would be appropriate at our full committee meeting that we have a response to that.

MR. KENNEDY: Okay.

MEMBER RAY: Are you talking about going from monthly to quarterly?

MEMBER STETKAR: I'm talking about going from monthly to quarterly on the diesels and from monthly to annually on the emergency sump pump -- I'm sorry, quarterly to annually on the emergency sump pump and perhaps no testing requirement for the shut-down cooling pumps.

MEMBER RAY: Okay. The question I was asking was going to having one diesel out indefinitely and just increasing the frequency of testing, which I'm not sure how the --

MEMBER STETKAR: Well, it's kind of covered under the thing I brought up, because the current tech specs existing today apparently say that if one diesel is out of service, you test the remaining diesel weekly.

MEMBER RAY: But forever, no limit on how long.

MEMBER STETKAR: I didn't look at that, and in the new tech specs, if one diesel is out, you test the remaining diesel monthly, once per month, which is, you know, an extension on the --

MEMBER RAY: There is an assumption -- again, I had this very brief discussion during the break. There is an assumption when you read this testing requirement that the diesel is required. I think the answer will turn out to be the diesel isn't required. It's superfluous, and so it doesn't matter what you do, and that becomes a part of the problem we have.

CHAIR SIEBER: But I'd like the staff to tell us that formally.

MEMBER RAY: Sure.

CHAIR SIEBER: And perhaps that's acceptable, perhaps not, and we'll determine that at the time, but the way it is right now, can't tell.

MR. KENNEDY: I'll provide clarification.

MEMBER RAY: I agree with that, Jack.

MEMBER BROWN: When you do the diesel testing, do you fully load them? For some --

CHAIR SIEBER: No.

MEMBER BROWN: Once a year.

CHAIR SIEBER: No, once a year you fully load.

MEMBER BROWN: You mean you start it up and just do it unloaded, and so you let them crap up by doing it once every week. If you don't load them for a while, they start carbonizing.

MEMBER MAYNARD: I don't know how they do it. Typically you do put some load. You just don't do a fully load, but you typically run it with a load versus not.

MEMBER BROWN: I know. You put it in a -- you load it enough to keep it from crapping up.

CHAIR SIEBER: And you run it long enough so that you aren't building up water, but that's practice as opposed to requirement.

MEMBER BROWN: I know we have not started it. We learned that lesson the hard way, because we started them up, loaded them a little, didn't run them long enough, went to start them, you know, after some period of time. Now they didn't start, so that may have been practice, but it didn't work, and we had to change the practice and made it a requirement.

CHAIR SIEBER: Okay.

MR. KENNEDY: I'll make sure to have a response to your concerns on that.

CHAIR SIEBER: Thank you. Relaxations of surveillance requirements, I think we need to know what they are, and they need to be justified. Okay, the conclusions, I think, are a repeat of what's in the SAR.

MR. KENNEDY: That is correct.

CHAIR SIEBER: So we can read these?

MR. KENNEDY: Yes. They are presented here to be once again on the record.

CHAIR SIEBER: Once you hand them your slides, they're on the record.

MR. KENNEDY: Okay. I'll just briefly talk about the open item, and that is the regulations say that a requalification program shall be conducted in a period not to exceed 24 months, and the licensee's program also says 24 months with the provision that they could extend it to 30 months on an irregular basis as long as they keep the average 24 months, and those aren't -- those don't line up, so we're addressing that through the REI process, and it should be resolved in March, before the full committee meeting.

CHAIR SIEBER: Okay. Well, thank you very much. We have five minutes left, and usually in that time I'd like to address a number of items. First of all, we recognize that there is one open item, and before we meet again that item should be closed.

In addition, we've asked a number of questions. Members have asked those, and I see people taking notes. The transcript will be available, and the questions that we asked that were not answered thoroughly should be noted and answered at the full committee meeting, and hopefully we'll have enough time to do that, because there's a number of questions.

If there is a way under the FICA rules that I could have an advanced copy of that, I write our final report, and I need information before the full committee meeting in order to be able to write a report that's consistent with what's going to be said. That's not a requirement, and it may not be allowed by the rules, because there has to be a way to get it into the public record. On the other hand, if it can be done, I would appreciate that.

The final meeting depends on the resolution of this open item. You know, we are not going to be the arbiter of a dispute over an open item.

At this point in the process of license renewal, the ACRS has an option if we find significant flaws that must be addressed to write an interim letter to the staff explaining what our objections are and giving them formal notice that we have such an objection before the final determination is made, so my question to each member as we go around the table is, in your opinion, is an interim letter required or not, and I'll start with John, since he's -- do you think we need an interim letter on any issue?

MEMBER STETKAR: I don't think so. I think that some of the issues that we brought up, depending on how they're resolved between now and the full committee meeting, you know, may affect our decision at that time, but --

CHAIR SIEBER: That's true.

MEMBER STETKAR: -- in terms of -- in terms of an interim letter on any particular issue, I would say no, in my opinion.

CHAIR SIEBER: Okay. Well, if we were to bring up an issue, it's an issue that either the staff or the applicant or both have missed or if there is a conflict between the applicant and the staff.

MEMBER ABDEL-KHALIK: Would the issues of monitoring of groundwater for treating releases and the adequacy of seismic analyses rise to that level?

CHAIR SIEBER: It could.

MEMBER SHACK: The tritium problem is not a safety issue. It might be an issue of a lot of things, but I don't think it's really a safety issue. The seismic walk-down is the one that I would be -- you know, as Dennis says, it seems to be indicated in the SRP, and, you know, since this was licensed back in the sixties when people didn't really concern themselves about that, that would be the one issue that would be the closest, certainly, to my point as being of some significance.

CHAIR SIEBER: Well, the question is, and I'll ask you, is that worth an interim letter?

MEMBER SHACK: I don't know.

CHAIR SIEBER: Well, it could be.

MEMBER SHACK: You know, I'm sort of tempted by the, you know, the informal statement we have that when you look at a very severe accident, even with no -- you know, as long as the confinement building is standing, you seem to meet 10 CFR 100, which is a pretty modest -- a lot of impact.

CHAIR SIEBER: Even considering failures.

MEMBER SHACK: Yes, so I guess I don't --

MEMBER STETKAR: The only consideration there is, of course, we aren't privy to what analysis, and seismic can be rather sinister in terms of what boundary conditions it violates.

MEMBER BLEY: And we, for specific reasons, we haven't seen the details of that analysis, so we don't know all the assumptions that were made in it. It strikes me, and I don't know the history of us writing interim letters, but this is an issue that could be very troublesome when you come back, so not getting it in a letter might be troublesome. I don't know the real answer.

CHAIR SIEBER: It might not be a bad idea if there is documents either possessed by the staff or the applicant that address this in more detail.

MEMBER SHACK: Well, we were told that there was no walk-down.

CHAIR SIEBER: Okay.

MEMBER BLEY: Or analysis of those, of the kinds of things that could cause two-over-one problems.

MEMBER BROWN: Well, why wouldn't -- I'm sorry. Go ahead, Jack.

CHAIR SIEBER: I think this is something that I would have to think over a little bit and perhaps consult with you all again. Sam?

MEMBER ARMIJO: Yes, I don't think an interim letter is needed with the exception of this issue of the seismic, but I'm a little concerned on a couple of things, and maybe it's I haven't found the right parts of the application or the SER -- is the spent fuel pool. I don't think I heard enough about it, how things are done. I think there's more risk in that area than in the operation of the reactor.

The other thing is the vessel embrittlement. I was surprised that there isn't any kind of a surveillance -- material surveillance program, because embrittlement is not just a function of fluence and temperature. There's also a time dependency, and, you know, getting your vessel properties from an accelerated high-flux -- accelerated irradiation experiments isn't exactly the same as having a good surveillance program. I don't know if there is anything that --

MEMBER SHACK: It's a little late now.

MEMBER ARMIJO: -- if there happens to be some component that's going to be taken out just to check it. That bothers me a lot, you know, that it's sort of just assumed that experiments done years ago will predict what's going on.

MEMBER SHACK: We can go up to Brookhaven and take some chunks.

MEMBER ARMIJO: Maybe somebody should, but, you know, even so, you know, you can damage the vessel and do a lot of damage to the core, and you still seem to have plenty of margins as far radiation release, so that's just troubling, but I'd like to hear more maybe in the full committee, or refer me to what's in the application, and I'll do some more reading on the pool, because there is an awful lot of fuel, apparently, in that pool, and I don't know how much analysis has gone into that.

MEMBER ABDEL-KHALIK: I guess there are five issues that -- and two of them have already been mentioned. First is the adequacy of the seismic analysis; two, the monitoring of groundwater for tritium releases; three, the relaxation of the surveillance requirements; four, the review process for in-core experiments; and five, the adequacy of natural circulation calculations.

CHAIR SIEBER: I think those are all legitimate. Dr. Ryan?

MEMBER RYAN: Said covered it well from my point of view, so my issues are covered. Thank you, though.

CHAIR SIEBER: Okay. Otto?

MEMBER MAYNARD: I don't believe we need an interim letter. I do think these are important issues. I think it's a heads-up for the next discussion. I think that, you know, Al's statement on the review that was done still stays within the Part 100 limits I think takes care of a number of the safety concerns.

I really think that what we might be talking about on some of these, it might be more of a generic issue than it is for a NIST-specific, and several of these things are things that we may need to have further discussions about whether it's generic or whether it's to this plant. It kind of gets into the adequacy of the review criteria, I think, but I do think specifically on the seismic they do need to come to the full committee meeting with both the staff and the applicant with a better discussion as to why it's not an issue or a problem.

MEMBER STETKAR: Otto, you mean generic in the sense of research reactors, not --

MEMBER MAYNARD: Research reactors, yes, because if we're going to be reviewing a number of others of these, I've got a feeling some of these same issues are going to come up.

MEMBER STETKAR: I just wanted to make sure it wasn't, you know --

MEMBER MAYNARD: And I do share the concern with the groundwater. That's not a reactor safety. I think it's more of a political issue than anything else, but it is something I think both the staff and the applicant need to -- I don't think we're doing -- I don't think there's adequate monitoring going on to really be able to detect it, but I don't think it's really a safety concern, either.

CHAIR SIEBER: And state law comes into play there, too, for discharges and environmental monitoring, which is not our jurisdiction. On the other hand, the NRC does have some jurisdiction over releases. Charlie?

MEMBER BROWN: I don't think there is an interim letter needed, either, rather than -- as long as we can get some more answers on the seismic issue. That was an interesting discussion. The one thing I'd like -- the redundancy issue is -- concerned me a little bit as we went through the report and the SER and stuff.

If you look at the radiation monitoring, and they've got, I guess, three channels as Harold pulled up the thing, and how are those -- how do they achieve the -- you know, if they're different detectors, different locations, how do you achieve the same redundancy in terms of having them perform whatever response is supposed to be performed? It'd be nice to know how that was achieved.

Other than that, it'd be nice to have -- I'd like to state that we got back here a minute ago on does it really matter at all if you -- even if you can't support or state any numbers, if you can say, yes, it's a factor of five below whatever the thing is or whatever, you know, that at least provides a calibration as to how close are you. I mean, just like the other numbers that were, you know, presented for the MHA.

CHAIR SIEBER: Okay. Ray?

MEMBER RAY: Well, I agree with Said's list, Jack. I'll defer to my experienced colleagues on whether the letter is a good idea or not. I'm a little mystified by the -- to me, the issue of groundwater as an unmonitored release path, I don't believe that's just an environmental question. I think pondering this issue, "Well, it's not a safety issue" -- well, I'm not sure what you mean by that, but anyway, on the letter question, you guys know best whether that's a good idea or not.

CHAIR SIEBER: I wish we did. Dennis?

MEMBER BLEY: Yes, the only thing I'd -- well, I'd compliment the staff and applicant on the presentations we had. The one thing I kind of urge staff to do is be a little careful about statements like, "The safety analysis doesn't require the ability to survive a second failure given the event that we're analyzing." I think if you go back and look you'll see under other criteria that's actually in place, and it doesn't send a great message.

CHAIR SIEBER: Yes, I guess the only satisfaction one gets is that you can do some pretty severe things to the fuel assembly and having nothing function and not have much of a dose impact, and so you have to ask the question is public health and safety protected in those instances, and under the current rules, it is, so how adequately one can argue that they meet the conclusions that are required by the law determines the extent to which each of these things is an issue.

Well, I appreciate the input from the members. I also agree with Said's list of issues, and I think that we ought to get them typed up, passed out to all of us, given to the applicant and the staff as our issues.

I think that we can add the question of redundancy at least to the extent of why is there just one DC power supply, and everything else I can sort of justify in my mind, but that one I struggle. It perhaps is too high a logical wall for me to scale today, and I need help to do that.

I think the application, though, however, is well prepared. I think that the folks from NIST that are here are well prepared and did a good job in presenting the application. I also think the staff has done a pretty thorough analysis and written a pretty good report, and I think the presentations were put together in a logical and consistent manner to provide a good record for this proceeding, and so I thank the applicant and the staff for their work, and I thank the members for their active participation and review of the material. With that, John Stetkar -- John, you're in charge of the next meeting.

MEMBER STETKAR: Dennis.

CHAIR SIEBER: Dennis?

MEMBER STETKAR: It's a common mistake, Jack.

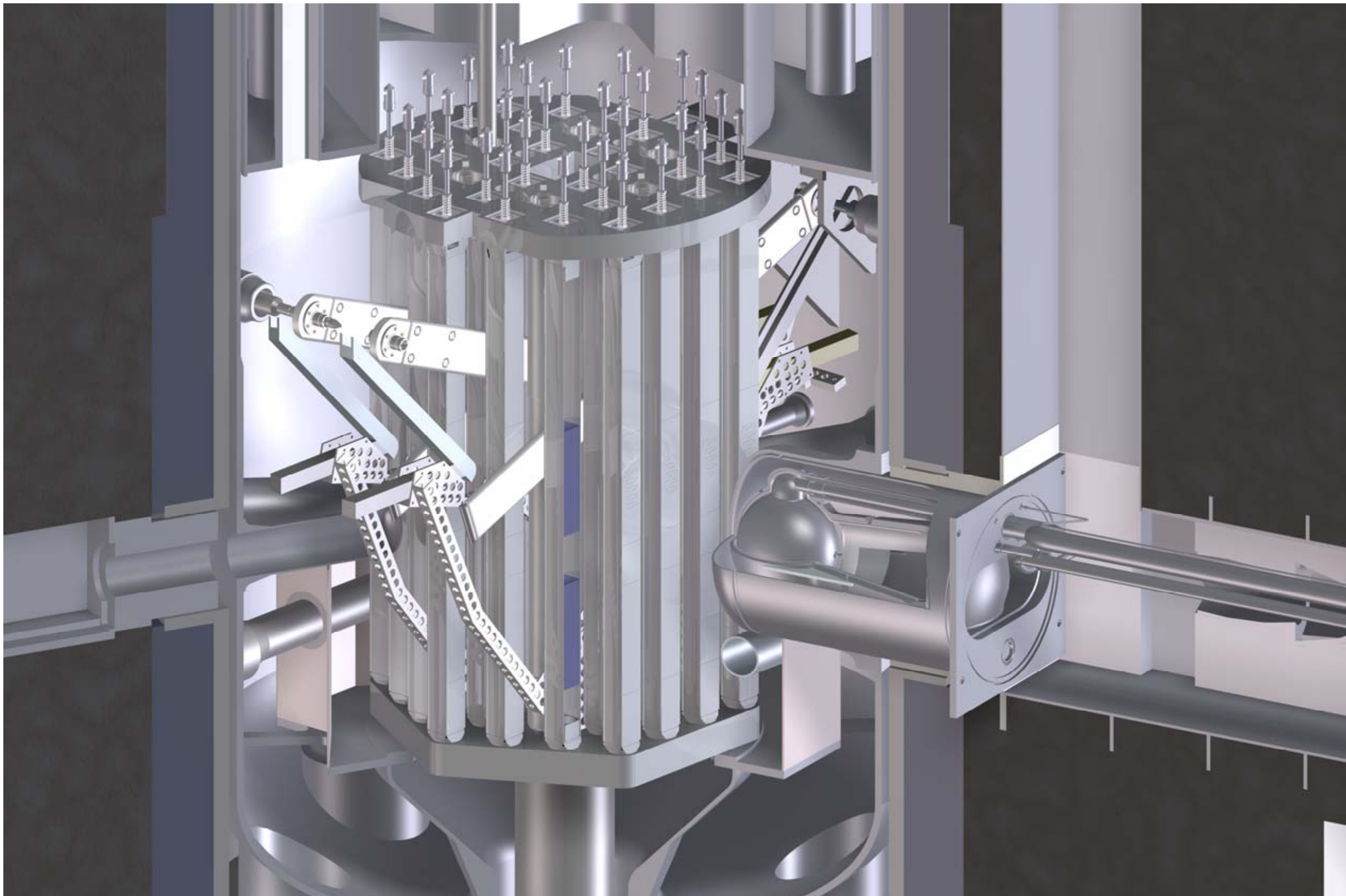
CHAIR SIEBER: Yes, I thought you were. The rule is you can never start early, but you can always start late, so you can decide when you're going to start the Beaver Valley meeting this afternoon.

MEMBER BLEY: Well, it's scheduled for 1:30.

CHAIR SIEBER: 1:30. That's perfect, and so with that, this meeting is adjourned.

(Whereupon, the foregoing matter was adjourned at 12:10 a.m.)

NIST Center for Neutron Research



ACRS Presentation (2/4/09)

- Background
- Operating history
- Reactor/major systems design, materials of construction and codes and standards
- Fuel element design and construction
- Expected high temperature fuel element integrity
- Engineered safety features and safety related electrical supplies
- Major modifications since last License renewal
- Fuel temperature coefficients
- Accident analyses
- Radiological impacts of normal and accident operations
- Baseline data of material condition
- Current surveillance programs
- Proposed Ageing Management of SSC's

The NIST Center for Neutron Research
A National User Facility

The mission of the NIST Center for Neutron Research is to assure the availability of neutron measurement capabilities to meet the needs of U.S. researchers from industry, university and other Government agencies.

- 23 instruments with access based on technical merit
- Highly interdisciplinary: basic/applied materials science & fundamental physics
- More than 2200 research participants per year
- Over 300 scientific publications per year
- Numerous partnerships with other agencies, industry, and academia (e.g. NSF, ExxonMobil, FDA, Smithsonian, Johns Hopkins, UMD,…)

Largest user program and highest productivity of any neutron facility in the US [1]

[1] The American Physical Society: *Access to Major International X-Ray and Neutron Facilities*, November 2008.

The National Context

“The NIST facility is the only U.S. facility which currently provides a broad range of world-class capability.” [1]

“…the NIST Center for Neutron Research (NCNR) currently has the largest number of users in the United States, largely because of its modern suite of cold neutron instruments.” [2]

“The highest priority for federal investments in neutron scattering is to fully exploit the best U.S. neutron source capabilities...for the benefit of the broadest scientific community.” [1]

“To improve access and to enable the user community to grow it is critically important to increase the number of beamlines and instruments at major facilities in the US.” [2]

[1] The Office of Science and Technology Policy Interagency Working Group on Neutron Science: *Report on the Status and Needs of Major Neutron Scattering Facilities and Instruments in the United States*, June 2002.

[2] The American Physical Society: *Access to Major International X-Ray and Neutron Facilities*, November 2008.

Strong Management Support
NIST and DoC

NIST and DoC leadership have been extremely supportive of and remain committed to the safe, effective, and reliable operation of the NIST Reactor as a critical component of the NIST measurement mission.

Past

- Upgrade from 10 MW to 20 MW (1984)
- Cold neutron source/guide hall (1994)
- Upgraded cold source installed (2002)
- Cooling tower w/plume abatement installed (2002)
- Initiative to expand access by supporting more instruments and developing new capabilities (2004)

Present/future

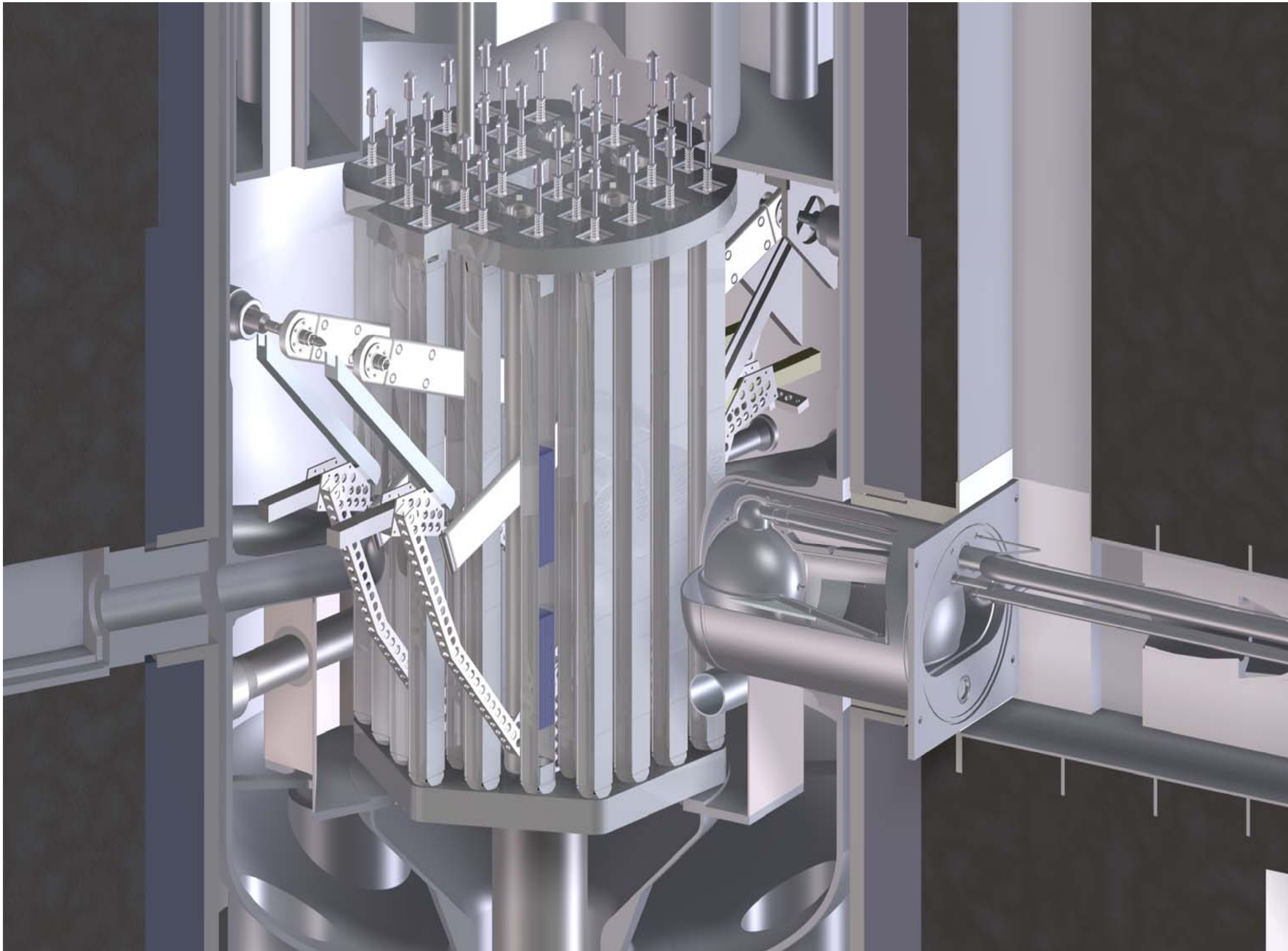
- NCNR Expansion and Reliability Enhancements Project (2007-2011)

OPERATING HISTORY

1984-1985	Increase to 20 MW, New D_2O.
1987	Change Shims
1994-1995	Guide Hall, Cold Source, Change Shims, New D_2O, New HX
2000	Change Shims
2004	Renewal Submitted, Change Shims, New D_2O
2008	Change Shims
1995-2008	Annual median full power days = 247

Reactor/ Major Systems Design Materials of Construction and Codes and Standards

- **Primary Heat Exchangers** were built to ASME B & PV Codes, Section VIII. Plate & Frame type design with primary flow through welded plate cassettes and Secondary Flow between cassettes held by gaskets.
- **Primary Piping, Pipe Fittings and Valves** met the requirements of ASME/ANSI B 31.3. All pipe fittings are forged 6061 T6 aluminum. Certified welders make all welds. Welds are radiographed and accepted on the basis of ASME Codes.
- **D_2O Main Circulating Pumps**. Single stage centrifugal units.
- Confinement Building was designed to meet the building Officials and Code Administrator (BOCA) Codes for the area. Building will withstand a 0.1g earthquake.
- **Materials of construction** - Reinforcing steel specified for the NBSR building conformed to Federal Specification in place at the time. (i.e. ASTM-A-305,ACI-613, Table 2 and ACI-318, Method 2, etc.) The confinement building is a reinforced concrete structure on a driven steel pile foundation.



Fuel Element Design and Construction

- **FE geometry unchanged since 1984 license renewal**
- **17 plates in upper and lower cores, 7-inch gap provides a thermal flux trap for beam tubes**
- **MTR fuel plates have a 50-year history of reliable use in many facilities**
- **<u>One Change:</u> U_3O_8 + Al dispersion fuel since 1991, when the fresh fuel loading was increased from 300 to 350 g of ^{235}U**
- **30 FEs: 38-day cycle with 4 FEs removed, 26 relocated, average burnup = 69%**
- **Roughly 7.4 kg of ^{235}U in SU core, 6.4 kg EOC**
- **HEU fuel plates and FEs produced by BWXT**

High-Temperature Fuel Element Integrity

- **Safety Limit:** **Al Cladding Temperature < 450 ^oC (threshold for blistering)**
- **SL assured if DNB* or OFI** never occurs**
- **LSSS:** **Power, D_2O Flow and Temperature Limits provide ample margins against DNB, OFI**
- **MCNP used to calculate FE power distribution, as well as axial, lateral, and plate-wise variations**
- **Average heat flux = 57 W/cm^2, peak is a factor of 3 higher**
- **Hot spots always in the outer-most plates, at the edges adjacent to the unfueled gap**
- **Normal operations: MCHFR=2.7, T_{max}=390 K (fuel)**

*DNB – Departure from Nucleate Boiling

**OFI - Onset of Flow Instability

ESF and Emergency Power

- Emergency Cooling System
- Confinement Building & Ventilation

- Redundant Building Feeders (2 of 3)
- Diesel Generators
- Station Battery

- UPS with AC and DC sources for critical loads, e.g. nuclear instrumentation
- AC and DC motors for Emergency Fans

MAJOR MODIFICATIONS
Safety and/or Scope

- Technical Specification Changes for Additions: Confinement Building Penetration Isolations for Guide Hall, Plate & Frame Heat Exchangers, Fuel Plate Material
- Cryostat installation
- NI Replacement
- Fuel loading and type
- Switchboards, battery & UPS
- Plume abatement cooling tower

Reactor Physics Parameters

- Coefficients of Reactivity are Negative:
  - Moderator* Temperature Coefficient -0.025 $\%\Delta\rho$ / ^{o}C
  - Moderator* Void Coefficient -0.030 $\%\Delta\rho$ / liter
  - Fuel Temperature Coefficient** -1.6e-4 $\%\Delta\rho$ / ^{o}C
- Prompt Neutron Lifetime = 800 µsec
- Delayed Neutron Fraction = 0.00757

* **Minimum Values** (The MTC and void coefficient depend on location and core life.)

** Doppler Effect from ^{238}U is very small in HEU fuel.

Credible Accident Analysis

- Credible accidents analyzed
  - Reactivity Insertion
  - Loss of primary coolant
  - Loss of flow
  - Improper fuel loading
- None result in fuel damage
- Loss of coolant results in ^3H releases resulting in public doses far below 10CFR20 limits

Maximum Hypothetical Accident

- Assumes <u>complete</u> flow blockage to one element
  - No credible initiating sequence
  - Complete instantaneous melting of all of the plates in that element
  - Immediate release of entire fission product inventory to reactor vessel
- Element assumed to have maximum inventory as determined by ORIGEN2

MHA (Cont)

- Gaseous fission products released from vessel to confinement building at rates corresponding to measured rates for ^{3}H
- All emergency ventilation systems assumed to function
- Standard codes (HotSpot, CAP88) are used to calculate public and staff doses

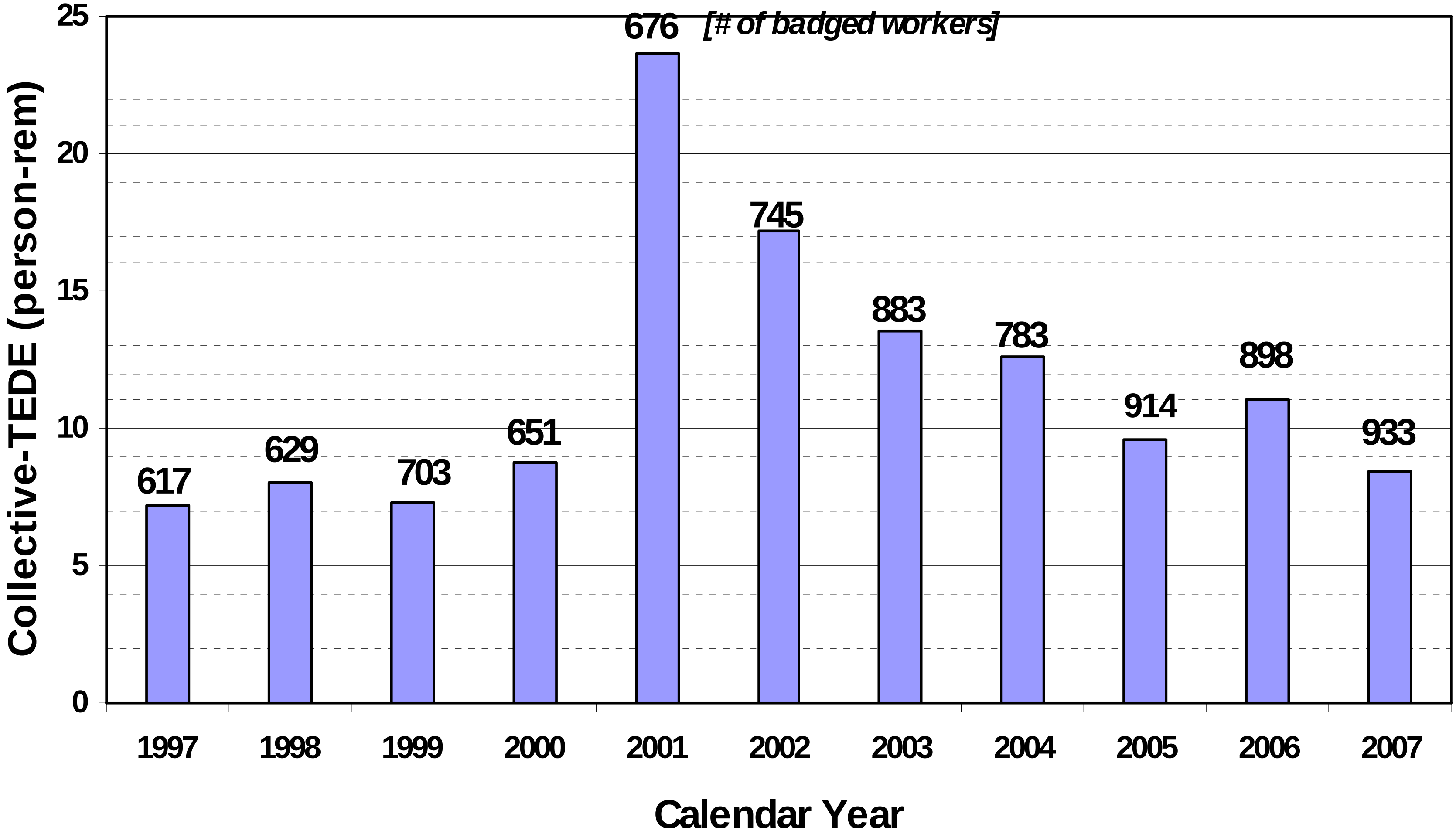
MHA Dose

Location General Public	Type of Dose	Dose (mrem)
Boundary (400m)	TEDE	7.0
	Thyroid	0.1

Location Staff maximum dose	Type of Dose	Dose in 10 min (rem)*
Operations level	TEDE	4.1
	Thyroid	0.02

* Note unit change

NCNR Personnel Exposure Summary
TEDE



Baseline Materials Data

- Reactor vessel will remain within established operating parameters for irradiation-induced loss of ductility beyond 2030
- Visual inspection of vessel internals revealed little to no evidence of corrosion
- In-house ultrasonic testing of primary piping revealed no significant wall thinning.

SURVEILLANCE PROGRAM

- Reactor Safety Operability
  ANSI/ANS-15.1-2007

- Non-safety, Aging SSC Operability
  System Expertise & Spending Plans

End

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Subcommittee

National Institute of Standards and Technology National Bureau of Standards Test Reactor License Renewal

February 4, 2009

William B. Kennedy, Project Manager
Office of Nuclear Reactor Regulation

Introduction

- National Institute of Standards and Technology (NIST) National Bureau of Standards Reactor (NBSR) License Renewal
- Topics for Now:
  – Licensing History
  – Current Licensing Status
  – Staff Review Criteria

Introduction

- Topics for Later:
  - Safety Evaluation Report overview
  - Results of application of the Standard Review Plan
  - Staff inspection history
  - Major issues
  - Staff conclusions

Construction Permit History

- 1961
  - The National Bureau of Standards applied for a construction permit and operating license for a 10 Megawatt thermal (MW(t)) heavy-water-cooled-and-moderated reactor
- 1963
  - The ACRS believed that, “the proposed reactor can be constructed at the Gaithersburg site with reasonable assurance that it can be operated without undue risk to the health and safety of the public,”
  - The Atomic Energy Commission (AEC) issued a construction permit

Operating License History

- 1967:
  - The opinion of the ACRS was that, “the reactor can be operated as proposed without undue risk to the health and safety of the public”
  - The Atomic Safety and Licensing Board concurred with the recommendations of the ACRS and regulatory staff
  - The AEC issued Provisional Operating License No. TR-5
  - The NBSR achieved first criticality December 7th

Operating License History

- 1969:
  - Full-power operation at 10 MW(t) began February 6
- 1970:
  - The ACRS reaffirmed its previous conclusion, and recommended, “conversion of the current provisional operating license to a full-term operating license”
  - The AEC issued Facility License No. TR-5 with a term of 15 years

Previous License Renewal

- 1980:
  - The National Bureau of Standards applied for a 20-year renewal and an increase in the maximum licensed power level to 20 MW(t)
- 1984:
  - The ACRS believed, “there is reasonable assurance that the renewal of the license... may be granted without involving any undue risk to the health and safety of the public”
  - The NRC issued the renewed license at the increased power level for a period of 20 years

Recent Licensing History

- Since the 1984 license renewal, NRC issued 3 license amendments:
  - change to guide tube isolation valve Technical Specification (TS) requirements for reactor operation
  - change in the type and number of primary heat exchangers allowed by the TSs, and a change in the designation of the National Bureau of Standards to the National Institute of Standards and Technology
  - administrative changes to the TSs

Status of Current License

- NIST currently operates the NBSR under the provisions of 10 CFR 2.109, “Effect of timely renewal application”
- NIST filed an application for license renewal April 9, 2004
- Facility Operating License No. TR-5 would have expired May 16, 2004

License Renewal Status

- NRC issued NUREG-1873, "Environmental Impact Statement for License Renewal of the National Bureau of Standards Reactor," January 2008
- NRC completed the Draft Safety Evaluation Report related to license renewal January 2009
  - one open item related to a timing requirement for periodic completion of the operator requalification program

Regulatory Review Criteria

- In accordance with Section 104 of the Atomic Energy Act of 1954, as amended (the Act), the NRC must "impose the minimum amount of regulation consistent with its obligations under this Act..."
- As defined by 10 CFR 50.2, the NBSR is a testing facility (reactor power greater than 10 MW(t)), a subset of non-power reactors
- 10 CFR Part 54 does not apply to license renewal for non-power reactors

Regulatory Review Criteria

- Non-power reactor license renewal is primarily conducted in accordance with 10 CFR Parts 20, 50, and 51
- The NBSR, as a test reactor, is also subject to the requirements of 10 CFR Part 100, “Reactor Site Criteria,” and review by the ACRS

Staff Review Guidance

- NUREG-1537, Part II, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” dated February 1996, provides the staff with review criteria
- NUREG-1537 was designed to apply to all non-power reactors, so all of the review criteria do not apply to each non-power reactor under review

Staff Review Guidance

- NUREG-1537 references other documents used in the review of non-power reactor licensing:
  - NUREGs pertinent to special areas of the review, e.g., emergency planning
  - Regulatory Guides, division 2
  - American National Standards Institute/American Nuclear Society ANSI/ANS-15 series standards

Staff Review Criteria

- Specific Criteria (chosen for greatest safety significance):
  - fuel cladding temperature does not exceed the safety limit anywhere in the core during normal operation and credible accidents
  - no radiation doses or releases of radioactive material exceed the limits specified in 10 CFR Part 20 during normal operation
  - calculated radiation doses from the maximum hypothetical accident are a small fraction of the dose guidelines specified in 10 CFR Part 100

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Subcommittee

National Institute of Standards and Technology National Bureau of Standards Test Reactor License Renewal

February 4, 2009

William B. Kennedy, Project Manager

Office of Nuclear Reactor Regulation

Overview

- Overview of Topics to be Covered:
  – Overview of the staff's Safety Evaluation Report (SER)
  – Results of application of the Standard Review Plan (SRP) (i.e., NUREG-1537)
  – Staff inspection history (presented by Johnny Eads, Chief, Research and Test Reactors Branch B)
  – Major issues covered by the staff's review
  – Principal safety conclusions

Overview of the SER

- The staff conducted its safety review using the applicable guidance found in NUREG-1537
- The staff based its review on information contained in the renewal application, as supplemented:
  - Safety Analysis Report (SAR)
  - Technical Specifications (TSs) proposed by the licensee
  - License responses to staff requests for additional information (RAIs)

Overview of the SER

- The staff used additional information sources during its review:
  - Annual reports submitted to the NRC (2000-2007)
  - First-hand observations
  - Inspection reports
- The staff used technical evaluation input related to the SAR, TSs, and RAIs provided by Washington Safety Management Solutions, LLC, under contract to the NRC

Overview of the SER

- Following the SRP and established NRC procedures, the SER covers a similar range of areas reviewed in an initial application for a facility license
- Areas of Review:
  - Site characteristics
  - Design of structures, systems, and components (SSCs)
  - Reactor core and control element designs
  - Nuclear design

Overview of the SER

- Areas of Review (cont.):
  - Thermal-hydraulic design
  - Coolant systems
  - Engineered safety features
  - Instrumentation and control systems
  - Electrical systems
  - Auxiliary systems
  - Experimental facilities and programs

Overview of the SER

- Areas of Review (cont.):
  - Public and occupational radiation protection
  - Radioactive waste management
  - Conduct of operations
  - Emergency planning, security planning, and operator training and requalification programs
  - Maximum hypothetical accident
  - Postulated credible accidents

Overview of the SER

- Areas of Review (cont.):
  - Adequacy of TSs
  - Financial qualifications, including decommissioning planning
  - Prior use of reactor components

Application of the SRP

- The staff applied NUREG-1537 during its review of the renewal application, including supporting guidance and standards referenced in the SRP
- The SRP has been the basis for license renewal since its issuance in 1996
- Because the SRP covers all aspects of non-power reactor licensing, the staff applied the portions of the SRP consistent with the scope of the renewal review

Facility Siting Criteria

- Review included:
  - Geography
  - Demography
  - Nearby facilities that could impact the reactor
  - Meteorological data
  - Hydrology
  - Seismology

Facility Siting Criteria

- Findings:
  – The reactor is appropriately sited for this type of facility
  – Hazards related to the site are not expected to pose a significant threat to safe operation of the facility during the period of the renewed license

Review of SSC

- Review included:
  - Reactor confinement building
  - Fuel design
  - Core support structures
  - Reactivity control devices
  - Nuclear safety systems
  - Coolant systems
  - Engineered safety features
  - Auxiliary systems
  - Radiation monitoring systems

Review of SSC

- Findings:
  – The design bases for the SSCs remain valid
  – The SSCs can reasonably be expected to continue to provide for safe reactor operation and shutdown

Reactor Characteristics

- Review included:
  - Reactor control
  - Flux distribution
  - Power peaking
  - Moderator temperature and void coefficients
  - Transient behavior
  - Fuel temperature
  - Safety margins

Reactor Characteristics

- Findings:
  - The neutronic and thermal-hydraulic behavior of the reactor provides reasonable assurance that the reactor can be reliably operated
  - The safety margins are adequate to protect the safety limit under all conditions

Electrical Power Systems

- Review included:
  - Electrical power systems for normal operation
  - Emergency power systems
    - redundancy
    - loads supplied during loss of offsite power
    - duration of availability of emergency power
- Finding:
  - The electrical systems are adequate to provide for normal operation and maintain safe shutdown during a loss of offsite power

Experiment Program

- Review included:
  - Experimental facilities
  - Accidents involving experiments
  - Administrative controls for experiment review and approval
- Findings:
  - Experiments should not pose a significant risk to safe operation of the reactor
  - New experiments will be properly reviewed and approved before being implemented

Radiation Protection

- Review included:
  - Radiation sources
  - Effluents
  - Wastes
  - Personnel monitoring
  - Environmental monitoring
  - Administrative controls

Radiation Protection

- Findings:
  - The radiation protection program is sufficient to maintain radiation doses below regulatory limits and is consistent with as-low-as-reasonably-achievable principles
  - Appropriate controls are in place to prevent uncontrolled releases of radioactive material or releases of material in excess of regulatory limits
  - There is reasonable assurance that the licensee will properly handle and disposition radioactive wastes

Accident Analyses

- Review included:
  - Maximum hypothetical accident (MHA)
  - Reactivity insertion
  - Loss of coolant
  - Loss of coolant flow
    - pump seizure
    - throttling of coolant flow
    - loss of shutdown coolant pumps
  - Misloading of fuel

Accident Analyses

- Findings:
  - The consequences of the MHA bound the consequences of all other accidents, and the dose consequences of the MHA are a small fraction of the 10 CFR Part 100 guidelines
  - The limiting safety system settings provide adequate safety margins to protect the safety limit for all postulated credible accident scenarios

Technical Specifications

- Review included:
  - Safety limit
  - Limiting safety system settings
  - Limiting conditions for operation
  - Surveillance requirements
  - Design features
  - Administrative controls

Technical Specifications

- Findings:
  - The proposed TSs satisfy the regulatory requirements of 10 CFR 50.36
  - The proposed TSs provide reasonable assurance that the facility will be operated as analyzed in the NBSR SAR

Prior Use of Components

- Review included:
  - Fuel degradation
  - Reactor vessel embrittlement
  - Control rod aging
  - Surveillance requirements

Prior Use of Components

- Findings:
  - Prior use of reactor components will not increase the likelihood of accidents or cause unanalyzed accidents
  - Surveillance requirements specified in the TSs provide reasonable assurance of adequate monitoring for degradation of SSCs

Staff Inspection History

- NRC inspections are conducted at NIST twice each year per Inspection Manual Chapter 2545, "RTR Inspection Program"
- Major areas inspected include reactor operations and maintenance, radiation protection, effluent and environmental monitoring, reactor surveillance, review and audit program, design changes, emergency planning, and security.

Staff Inspection History

- In the last two years, the reactor inspection program has not identified any violations or non-conformances at the NIST reactor facility.
- Based on the inspection history, NIST has a very stable, well managed, safety conscious operating program. Recently, significant advances in their engineering program have been evident (new design change process implemented, drawing controls enhanced, new shield panels for ALARA installed).

Major Issues

- Revisions to the Facility TSs:
  - As part of a supplement to the renewal application, the licensee proposed revised TSs to better conform to the guidance in ANSI/ANS-15.1
  - The proposed TSs included significant technical changes to reflect updated analyses in the SAR
    - safety limit
    - limiting safety system settings
    - limiting conditions for operation
    - surveillance requirements

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Major Issues

- Revisions to the Facility TSs:
  - The staff spent considerable effort to ensure that the proposed TSs do not omit any safety-significant conservatism contained in the current TSs
  - The staff found inconsistencies between the proposed TSs and the guidance found in ANSI/ANS-15.1 and the requirements of the regulations
  - All inconsistencies were remedied in a manner that ensures public health and safety, conformance with applicable regulations, and allows the licensee to operate the facility as intended

Major Issues

- Open Item: Operator Requalification Program
  - 10 CFR 55.59(a)(1) states, “this [requalification] program shall be conducted for a continuous period not to exceed 24 months in duration.”
  - The licensee’s program states, “the program shall be administered over a normal period of 24 months, to be followed by successive 24 month periods, with no period to exceed 30 months.”
  - The open item is being addressed using the RAI process and should be resolved March of 2009

Safety Conclusions

Based on its safety evaluation, the staff concludes:

- The design, testing, and performance of SSCs important to safety during normal operation are acceptable; safe operation can reasonably be expected to continue
- The licensee’s management organization is acceptable to maintain and safely operate the reactor

Safety Conclusions

- The licensee’s research activities and programs, including experiment malfunctions, will not pose a significant risk to continued safe operation of the facility
- Exposures from and releases of radioactive effluents and waste from the facility are not expected to result in doses or concentrations in excess of the limits specified in 10 CFR Part 20, and are consistent with as-low-as-reasonably-achievable principles

Safety Conclusions

- The licensee has conservatively considered the consequences of a bounding maximum hypothetical accident and shown the radiological consequences to be a small fraction of those specified in 10 CFR Part 100
- The licensee has conservatively considered an appropriate range of postulated credible accidents using appropriate initiating and mitigating assumptions

Safety Conclusions

- The renewed Facility Operating License and TSs provide reasonable assurance that the licensee will operate the facility in accordance with the assumptions in the SAR
- No significant degradation of SSCs has occurred, and the TSs will continue to provide reasonable assurance that no significant degradation of SSCs will occur

Safety Conclusions

- The licensee’s physical security plan continues to be acceptable to protect its special nuclear material
- The licensee’s emergency plan provides acceptable assurance that the licensee will continue to be prepared to assess and respond to emergency events

Safety Conclusions

- Continued operation of the NBSR during the period of the renewed license poses no significant radiological risk to the health and safety of the public, facility personnel, or the environment

Reactor Vessel Embrittlement

- SRP Guidance:
  - Acceptance criterion:
    - The licensee’s analysis should show that unacceptable levels of deterioration will not be reached during the license period
  - Review procedures:
    - The reviewer should study Chapter 13, “Accident Analyses,” of the SAR to determine if the applicant has chosen proper components and systems for consideration
    - The reviewer can consider the performance of similar components in reactors or environments comparable to the facility under consideration

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Reactor Vessel Embrittlement

- NRC Review:
  - Acceptance criterion:
    - The licensee’s analysis shows that the potential vessel embrittlement during the license period will not cause failure of the reactor vessel
  - Review procedures:
    - The staff reviewed the licensee’s accident analyses and found that the reactor vessel is a proper component for consideration
    - The staff evaluated the applicability of the references cited in the licensee’s analysis and found them to be applicable

Reactor Vessel Embrittlement

- NRC Review (Cont.):
  - Review procedures (cont.):
    - The staff reviewed the licensee's assumptions and analysis and found them to be reasonable
  - Findings:
    - The licensee's analysis is conservative and shows that embrittlement due to neutron interactions with the vessel materials will not cause unacceptable deterioration of the reactor vessel
    - If the reactor vessel were to fail, the licensee has analyzed a complete loss of coolant and shown the effects to be bounded by the MHA