UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 6, 2009

LICENSEE: Entergy Nuclear Operations, Inc.

FACILITY: Indian Point Nuclear Generating Unit Nos. 2 and 3

SUBJECT: SUMMARY OF JUNE 15, 2009, MEETING WITH ENTERGY ON THE TRANSFER OF SPENT FUEL FROM UNIT 3 TO UNIT 2 AT INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 (TAC NOS. ME0408 AND ME0409)

On June 15, 2009, a Category 1 public meeting was held between the Nuclear Regulatory Commission (NRC) and representatives of Entergy Nuclear Operations, Inc. (licensee) and Holtec International, Inc. at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the licensee's plan to transfer spent fuel from the spent fuel pool at Indian Point Nuclear Generating Unit No. 3 (IP3) to the spent fuel pool at Indian Point Nuclear Generating Unit No. 2 (IP2), and from there to the independent spent fuel storage installation which already exists at the site.

The licensee presented information on the method they would use to transfer the spent fuel, using a specially designed canister and an existing shielding cask to move the spent fuel across the site within the protected area of the site. The licensee discussed their plan to submit license amendments for IP2 and IP3 to request NRC approval for this fuel transfer. A transcript of the meeting, excluding the discussion of proprietary information, is provided as Enclosure 1. A list of attendees is provided as Enclosure 2, but may not be all-inclusive. The NRC's handout is provided as Enclosure 3, and the licensee's handout as Enclosure 4.

No Public Meeting Feedback forms have been received. Please direct any inquiries to me at 301-415-2901, or by email to John.Boska@nrc.gov.

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. List of Attendees
2. Transcript
3. NRC Handout
4. Licensee Handout

cc w/encls: Distribution via Listserv

Enclosure 1

List of Attendees

UNITED STATES
NUCLEAR REGULATORY COMMISSION

MEETING ATTENDANCE FORM

Subject: Meeting with Entergy on Spent Fuel Transfer at Indian Point Nuclear Plant

Date: June 15, 2009 Location: NRC Headquarters

PLEASE PRINT ALL INFORMATION LEGIBLY

NAME	ORGANIZATION
Patric W. Conroy	Entergy - IPEC
Floyd W. Gumble	Entergy - IPEC
Joseph DeFrancesco	Entergy - IPEC
JOHN SKONIECZNY	ENTERGY - IPEC
MARK RUNION	ENTERGY - IPEC
Robert WALPOLE	ENTERGY - IPEC
ROGER WATERS	ENTERGY - IPEC
Tammy Morin	Holtec International
Edward J. Weinkam	Entergy
KENNETH A. PHY	HOLTEC
BRIAN GUTHERMAN	ACI
John McCann	Entergy
Andrew Barto	NRC/NMSS/SFST
DAVID TANG	NRC/NMSS/SFST
LARRY CAMPBELL	NRC/NMSS/SFST
STEFAN ANTON	HOLTEC INTERNATIONAL
DOUG PICKETT	NRC/NRR/DORL
Scott Burnell	OPA
Sarah Anderson	Exchange Monitor
Bob Nelson	NRC/NRR/DORL
Kent Wood	NRC/NRR/DSS/SRXB

John Boska NRC/NRR/DORL/LPL1-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION

MEETING ATTENDANCE FORM

Subject: Meeting with Entergy on Spent Fuel Transfer at Indian Point Nuclear Plant

Date: June 15, 2009 Location: NRC Headquarters

PLEASE PRINT ALL INFORMATION LEGIBLY

NAME	ORGANIZATION
Melanie White-Cyon	NY AREA
Zhian Li	NRC/SFST
Eric Benner	NRC/SFST
Mike Call	NRC/SFST
BOB TRIPATHI	NRC/SFST
GEORGE THOMAS	NRR/DE/EMCB
Kim Ropon	NMSS/SFST/CSDAB
John Parillo	NRR/DRA/AADB
Elizabeth Thompson	NRC/SFST
Luis Cruz	NRC/NMSS/SFST
Robert Taylor	NRC/NRR/AADB
Rob Temps	NRC/NMSS/SFST
DyLanne Duvigneaud	NRC/NRR/AADB
EARL LOVE	NRC/NMSS/SFST
Michail Waters	NRC/NMSS/SFST
Kim Green	NRC/DLR
NANCY SALGADO	NRR/ADRO/DIRS/IOLB
Steven Jones	NRR/DSS/SRPB
Zach Curran	NRO/DCIP/CCIB
Anthony Minarik	
Ram Pankhill	NRC/SFST

John Goshen NRC/NMSS/SFST/LB

Enclosure 2

Transcript

UNITED STATES NUCLEAR REGULATORY COMMISSION
BRIEFING ON INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
ENTERGY NUCLEAR OPERATIONS, INC.
+ + + + +
Monday
June 15, 2009
+ + + + +

The meeting convened at 1:00 p.m., John Boska, NRC's Indian Point Project Manager, Office of Nuclear Reactor Regulation, Division of Operating Reactor Licensing, presiding.

Entergy Nuclear Operations, Inc:
John McCann (Director Nuclear Safety Licensing)
Joe DeFrancesco (Project Manager)
Brian Gutherman (Licensing Consultant, ACI)
John Skonieczny (Senior Engineer)
Floyd Gumble (Supervisor Reactor Engineering)
Patric Conroy (Director Nuclear Safety Licensing)
Robert Walpole (Licensing Manager)
Edward Weinkam (Senior Manager Licensing)
Roger Waters (Licensing Engineer)
Mark Runion (Manager of Project Management)

Holtec International:
Kenneth Phy (Senior Project Manager)
Dr. Stefan Anton (Vice President Engineering)
Tammy Morin (Licensing Manager)

PROCEEDINGS

MR. BOSKA: All right, it's one o'clock. We are ready to start the meeting. My name is John Boska. I'm the NRC project manager for the Indian Point Nuclear Plant which is located in New York. Today we are meeting with Entergy, the licensee for Indian Point. We also have representatives from Holtec, a contractor for Entergy. Copies of the slides are available near the main entrance to this room if you don't have a copy of the slides.

There are NRC slides and Entergy slides. We also have meeting feedback forms in the same location as the copies of the slides if you would like to give us feedback on this meeting. The meeting feedback forms can be left in the box next to the feedback forms or they can be mailed to the NRC.

I would like to start by asking the NRC staff at the table to introduce themselves. I'm John Boska from the Office of Nuclear Reactor Regulation.

MR. NELSON: Bob Nelson, deputy director, Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulation.

MR. Taylor: Rob Taylor, Chief of the Accident Dose Branch in the Office of Nuclear Reactor Regulation.

MR. WOOD: Kent Wood, technical reviewer, Reactor Assistance Branch.

MR. BENNER: Eric Benner, Acting Deputy Director of the Licensing and Inspection Directorate in the Division of Spent Fuel Storage and Transportation.

MR. CAMPBELL: Larry Campbell, Branch Chief, Criticality Shielding and Dose Assessment Branch, Spent Fuel Storage and Transportation Division, NMSS.

MR. GOSHEN: John Goshen, NMSS project manager for Holtec HI-STORM

Systems.

MR. BOSKA: Next, I would like the Entergy people at the table to introduce themselves.

MR. WATERS: aRoger Waters, Indian Point Licensing

MR. WALPOLE: Bob Walpole, Licensing manger at Indian Point.

MR. CONROY: Pat Conroy, Director of Nuclear Safety Assurance for Indian Point.

MR. DEFRANCESCO: Joe DeFrancesco, project manager for Indian Point.

MR. ANTON: Stefan Anton, Vice President of Engineering, Holtech International.

MR. CONROY: I would like to point out that we have from corporate, from Entergy, we have John McCann, the Director of Fleet Licensing. We also have Ed Weinkam, the senior licensing manager for the northeast in addition to the Entergy and Holtec staff you had mentioned. Thank you.

MR. BOSKA: All right, next slide, please. We've done the introductions. We will move on the next slide. This is a category one meeting today. The purpose of today's meeting is to allow Entergy to discuss their proposal with the NRC staff. The public will be allowed to ask questions of the NRC staff during the question and answer period.

Next slide, please: Entergy is proposing to transfer spent fuel from the Unit 3 spent fuel pool to the Unit 2 spent fuel pool and then into the independent spent fuel storage installation which already exists at the site.

The second half of this meeting will be closed to the public to allow for the discussion of proprietary information.

Next slide, please: This is our agenda today. Next on the agenda is an overview of the project by Entergy. And with that, I'll turn the microphone over to Entergy.

MR. CONROY: Okay, thank you John. Good afternoon everybody. As John indicated, the objective today is for Entergy to present our plans to transfer the IP3 spent fuel to the IP2 spent fuel pool and from there into the dry cask storage facility at the existing ISFSI.

As we have discussed, we have been in conversation requesting an accelerated review associated with these plans once the license amendment requests are submitted, and to the extent that that is achievable. And we look forward and welcome and thank the NRC staff and everybody here for the opportunity to meet with you so that we can basically have an open exchange of information and answer as many questions that everybody might have.

Next slide please. Just an overview of the agenda that we have prepared. We have a presentation to go through essentially. We will give you an overview of the project as a whole. From there, we will get into a review of the fuel transfer equipment design and also, the operations process for that. We will then move into some of the license acceptance criteria and results.

And then finally, we will outline the licensing amendment requests that we believe are necessary for us to go forward with our plans that we will be submitting to the NRC by the end of June.

Finally, we'll have some closing remarks and then turn it back over to NRC for questions and answers. So with that, I'm going to turn it now over to Joe DeFrancesco who is the project manager for this project.

MR. DEFRANCESCO: Thank you Pat. The first thing we would like to do is just discuss some terms and definitions that we are going to be seeing throughout this presentation just so we are all on the same page here. And the FSB refers to the Indian Point fuel storage buildings, either Unit 2 or Unit 3. The STC is the shielded transfer canister. That is actually the canister that we are designing and planning on implementing for the transfer of the fuel. The Hi-Trac 100D is the licensed transfer cask. The STC or shielded Transfer Canister will actually be placed in the HI-TRAC for transportation.

The VCT is the vertical cask transporter. That is the cask transporter that is currently licensed under dry cask storage operations that we currently use at Indian Point. It's a crawler as we call it or a tracked vehicle, somewhat like a tank, if you will. That will be used as the transport vehicle for the Hi-Trac.

LPT is the low profile transporter and at Indian Point, we will be using Hilman rollers cart which is a specially designed cart for our dry cask storage operation which we have already used for Unit 2 and Unit 1 operations. And also, we will be using air pads. The air pads will actually be used when we come out of and go into Unit 3 fuel storage building. And the LCO is the basically technical specification Limiting Condition of Operation. That's the conditions we must apply for maintaining our license.

Next slide, please. I would like to give a little brief summary of how we got where we are right now. Currently, we are using dry cask storage to maintain our Unit 2 fuel inventory. To date, we've got 3 Unit 2 casks on our storage pad and we use a Holtech HI-STORM system.

When we were designing and implementing this we had -- one of the major challenges we had was the fuel storage building Unit 2 overhead crane.

The existing crane was a 40-ton capacity non single failure proof crane.

We did a feasibility study as far as upgrading that crane to 100 ton plus capacity. And what was concluded as part of that feasibility study was that the superstructure of that fuel storage building just could not handle that upgrade.

So we then looked into using a new crane, the gantry crane, a 110-ton plus crane. It was basically the most practical, because to modify the Unit 2 fuel storage building superstructure was just too enormous of a task. So we ended up installing this 100-ton gantry crane doing the cask loading at unit 2 as well as bringing the Unit 1 fuel over and doing the cask loading at Unit 2 also, for five Unit 1 casks.

So while we were doing the Unit 2, we started reviewing and trying to figure out what are we going to do with Unit 3, knowing we had certain limitations in Unit 3. So roughly the 2007 timeframe is when we really started to look into our options for Unit 3.

Since the Unit 3 fuel storage building is very similar in design to the Unit 2 fuel storage building and we had the same 40-ton capacity non-single-failure-proof crane, we knew we had to do something: And the same feasibility is applicable to the Unit 3 fuel storage building.

The member size that we'd have to modify that building was just too big a job to really implement upgrading that crane.

Secondly, looking at installing a gantry crane in the fuel storage building floor also had some significant challenges.

The first thing is the floor elevation in the fuel storage building in Unit 3 is approximately 23 feet lower than in the Unit 2 fuel storage building which would really cause an engineering challenge in trying to design a gantry crane to fit in that building.

The member size would again be a problem.

Secondly, in that building, the floor layout, the truck bay floor layout is very different in Unit 3 versus Unit 2.

In Unit 3, we actually have a liquid radwaste processing system in our truck bay area.

We don't have the same area to basically install a gantry crane.

So based on those issues we realized we really can't go the same way we went with Unit 2 by installing a gantry crane inside the fuel storage building.

So as a result, we felt that the best, most practical and most economical approach, and the safest approach was to basically do an interunit transfer of the fuel, taking the fuel from Unit 3 fuel storage building, and transporting it to Unit 2 fuel storage building where we then load it into the fuel pool and then do a cask loading, once all the fuel -- once the planned amount of fuel was put in there.

So currently we have three casks from Unit 2 loaded on the pad and five from Unit

1 actually loaded on the pad.

So we have successfully been doing dry cask storage.

We also have a plan to continue dry cask storage for Unit 2 fuel starting very shortly.

We will be doing some additional casks this year.

Now our Unit 3 has lost our full core off-load capability as a result of our last refueling outage which ended in the spring. So we no longer have full core discharge capability at Unit 3.

The crane upgrade issue we discussed already -- it's not feasible to upgrade the crane and the gantry crane will not fit.

Our project scope, basically what our project is doing is we are going to replace our 40-ton overhead fuel storage building Unit 3 crane in accordance with NUREG 0544 to make it a single failure proof crane.

The Unit 3 fuel storage building truck bay door will be enlarged to accommodate the size of the components we are going to be using, the shielded transfer canister and the Hi-Trac going in and out.

In addition as a result of that, we have some interferences such as electrical conduits and piping that needs to be rerouted. We are going to level the truck bay floor and that basically ensures we have the stability when moving the Hi-Trac in and out of the building. We will be using the air pads as a low profile transporter for Unit 3 and leveling ensures our capability of maintaining stability and we used air pads when we did Unit 1, when we took the fuel out of Unit 1 using the Hi-Trac.

Roadway haul improvements is a significant issue that we are evaluating.

We are looking to ensure that the roadway will meet all stability requirements for transporting the Hi-Trac with the vertical cask transporter. We are going to ensure that vertical cask transporter stability is maintained at all times during this trip along the haul path.

We are going to be doing ground penetrating radar to investigate what the conditions are underneath the surface, soil compaction studies.

We are going to evaluate any of the undergrounds that are in place and will harden them, modify them as necessary to ensure they can withstand the pressure that the vertical cask transporter with the loaded Hi-Trac will exhibit.

The resulting roadway -- just on that note, the resulting roadway pressures of the VCT carrying a Hi-Trac is similar to that of a tractor/trailer. And our roadways are designed currently for H-20 loads which handles that tractor/trailer.

The other thing of note is Indian Point is basically built on rock, solid rock, and that is a very significant obviously structural base. And we had our issues with that when we were doing the installation for Unit 2.

Also about half of the travel path that we will be using has already been evaluated and in fact used as part of the Unit 2 and Unit 1 dry cask storage operations that we have currently completed and will be doing more.

And one modification we know we are doing, is we are going to actually put concrete runways in the roadway that will match the configuration of the vertical cask transporter track so that the vertical transporter will actually run on those concrete runways. The protected area boundary fence will be relocated to ensure that the fuel transfer will always be maintained inside the protected area.

Currently, it's not, but for this project it is being relocated for reasons, but this fence will be relocated so that the transfer is always inside the protected area.

The fuel transfer equipment we will be using is the shielded transfer canister which is a new piece of equipment.

Other pieces are what we currently use for dry cask storage operations, the Hi-Trac, the vertical cast transporter and the low profile transporters.

The rest of the scope is to transfer the Unit 3 assemblies to Unit 2 and basically that's going to happen by the shielded transfer canister will be placed inside the Hi-Trac which will be then secured, attached to the vertical cask transporter, and moved over to the Unit 2 fuel storage building.

Next slide. This just shows a sketch of our fuel storage building, and you could see, that's where the Hi-Trac is, and you will slide, come back down and that will be the shielded transfer canister, it will then be moved over to the pool, to the cask loading area and that's where the shielded cask transfer canister will be loaded with the fuel.

Next slide. Oh, I'm sorry. Go back.

The area just to my left, that area, that's where the liquid radwaste processing system is and that's why we really can't install a gantry crane there. There just isn't enough floor space in there without some major modifications to that plant which is really not feasible and really not economical.

Next slide: This slide, I just tried to indicate the elevation difference between the Unit 2 and Unit 3. You could see that the elevation of the Unit 3 fuel storage building is at 55-foot, floor elevation, 55-foot and it goes up to 95-foot.

Unit 2 is at 77-foot, six inches.

So it is a significant difference in elevation which would end up resulting in that significant design change and member size for a gantry crane, 110-ton gantry crane.

Okay. Next slide. The fuel transfer equipment, like I said, that we plan on using is the new shielded transfer canister and that's going to hold up to 12 fuel assemblies for transfer. The existing HI-Trac with a solid top lid, and this is again, like I said, we have been using the HI-Trac as part of our dry cask storage operations.

Special lifting devices are all part of this and they are all going to be designed to ANSI N14.6. The low profile transporters we talked about, the Hilman roller cart specially designed for dry cask storage.

For Unit 2, we will be using that to transport the HI-Trac with the shielded transfer canister on it into the Unit 2 fuel storage building and then air pads will be used to transport the Hi-Trac and shielded transfer canister outside of the Unit 3 fuel storage building.

And the existing vertical cask transporter, again, that is the transport vehicle that we use to carry the HI-Trac on the transport roadway.

Some basic design features for the shielded transfer canister. Its approximate dimensions are about 52 inches in diameter and about 183 inches high, roughly four foot in diameter and a little bit over 15-foot high.

Lifting attachments, we have trunnions for lift yoke.

The handling weight loaded will be less than 40 tons including the rigging that is going to be part of the operation.

The capacity is up to 12 fuel assemblies, and the fuel basket design is a stainless steel frame with Metamic as the absorber.

The lid design is a bolted flat head with an elastomer seal.

Safety features include two relief valves and maintenance-wise we are going to be required to do calibration for the relief valves, trunnion inspections and then general inspections such as for the seal to be sure it is not cracked, in good shape, things like that.

And the codes and standards is the ASME Boiler and Pressure Vessel Code as shown there. And ANSI N14.6.

The HI-Trac is currently what we use now for our dry cask storage. It's a 100 day.

It will have a solid top lid with lid closure bolting and elastomer seal.

The maintenance requirements per Part 72 FSAR and general inspections, again, similar with the seal inspections for cracking. And the codes and standards ASME Boiler and Pressure Vessel Code Section III, Subsection ND, and ANSI N14.6.

Next slide: Just want to go through a quick summary of basically how the sequence of the operations will happen.

This of course will be performed in accordance with all approved administrative controls and procedures.

First step is to move the shielded transfer canister into the Unit 3 spent fuel pool.

I showed you on the sketch before how that was going to go in.

Then we will load the fuel and install a lid while it is in the spent fuel pool.

Then once the lid is installed, we will remove the shielded transfer canister from the pool and place it in the Hi-Trac.

We will remove water from the shielded transfer canister for an air space and install lid bolting.

Now that air space is approximately eight inches.

Using the low profile transporter, in this case we will be using air pads.

We will move the HI-Trac with the shielded transfer canister inside of it outside of the Unit 3 fuel storage building.

The HI-Trac will then be moved with the VCT.

It will be secured on the VCT, the vertical cask transporter.

And the vertical cask transporter will then move the HI-Trac with the STC in it along the haul path to the Unit 2 fuel storage building where we will go to our existing LPT, which is right outside of our Unit 2 fuel storage building, in what we call the alleyway into the Unit 2 fuel storage building and we have our Hilman roller cart there where the vertical cask transporter will lower the HI-Trac onto the Hilman roller cart and that will then be rolled into the Unit 2 fuel storage building.

Then using the Unit 2 gantry crane, we'll lift the shielded transfer canister out of the HI-Trac and place it in the spent fuel pool cask handling area.

There we would remove the lid and we move fuel into the Unit 2 racks.

This is an overhead trying to show the haul path for the unit -- there is the Unit 2 fuel storage building -- I'm sorry, Unit 3 fuel storage building.

We will be coming out, following the haul -- again, it may not show it here but it will be all done inside the protected area.

That modification will be performed before we do this work.

We will track along the haul path, all around and then we will roll into there (indicating) and then go up to the right, and keep going up, and then we will turn and that's -- we will end up in the alleyway to the Unit 2 fuel storage building and that's the area of the LPT.

And that's Unit 2 fuel storage building.

One area I want to highlight is back where we come in, where the existing security gate is now.

The area just to the right of that area where the road is divided, that area's already been used. That area's already been used as part of Unit 1 dry cask storage, and so it's already -- the vertical cask transporter has already been used on that and successfully used and also obviously from Unit 2, the haul path from Unit 2 fuel storage building out to our pad has been used also. That's it.

With that, I would like to turn over to Bob Walpole, for a licensing discussion.

MR. WAPOLE: Thank you. I'm Bob Walpole, the licensing manager at Indian Point.

I just want to give you a general overview of what you will be receiving from us in the next two weeks.

We're going to go into a little later in the presentation after we go through the specifics and the details as to how we are designing and how we are analyzing this, the specific technical specifications, design features and also the operating license changes that we will be making.

But what we have prepared for you with our teammate Holtec is a very detailed report we have been working together on working on the specifics of the entire project and we have extracted from that portions to include our 50.90 analyses.

And we in the next two weeks will be submitting that to you.

I would like to thank you for your letter of June 11.

We have been going through that over the past several days making sure that all the analyses and all the studies that we have been doing are responsive to the information that you presented in there.

So I appreciate you working with us. It's making this project useful to us, so thank you for that.

Thank you for that.

By saying in the next two weeks we will be submitting this and we are going to be asking for an approval date of August of 2010.

That will support our initial package -- first plan of making moves in the October 2010 timeframe.

Our goal is to give you a package that is -- goes through sufficiently and is able to be accepted and used by the NRC and also give you a package that you won't have any additional questions on.

And by having these public meetings with us I think that helps us understand what your issues are in making sure that when we submit something to you it's all inclusive and it's fully responsive.

As I say, later on we will give you a discussion of the technical specifications, its design features and operating license in this package today that we are talking about.

And I would like to turn it over to Stefan Anton of Holtec, Vice President of engineering

MR. ANTON: Yes, Good afternoon, this is Stefan Anton and I plan to give a brief overview of the analysis that had been performed as part of the system and to -- okay, there is the first slide

-- over the analysis and the result of the preliminary result of the analysis.

I have divided my -- this overview into three sections. I hope it makes it a little easier to understand.

First I will just talk about the acceptance criteria that we have been applying to these analyses.

This is -- there is nothing really unusual in there, but we will -- we want to go through that first.

The second section would be a brief description of the actual calculations that were performed and present the principal results. And there was a telephone conference in March and in direct preparation for that conference there were already slides that had been submitted, which gave more details about the methodology of the analysis.

I'm not going to repeat that here in detail.

I will just briefly talk about these. And finally the third part, I will go through an overview of all the accidents that have been included in the analysis. So let's start with the acceptance criteria. The next slide, please.

For the criticality analysis, we've been performing the criticality analysis consistent with the 10CFR 50.68 which is the methodology that we have been using, that we are very familiar with for the spent fuel storage applications over many, many years.

50.68 gives you essentially two options in respect to the treatment of soluble boron and we have been choosing the more conservative one under the normal condition. We already consider the loss of all boraflex, so under normal conditions we analyze that with pure water where as for the presumed accident conditions, we then take credit for the soluble boron, that's considered with the spent fuel analysis.

Next slide, please.

This is just an overview of the various regulations and guidance's that are relevant for the criticality evaluation. There is nothing surprising in there, and that's the same on the next slide, the various parts including the famous Kopp memo that are listed out here.

We have been taking into account in the analysis the recent discussions that have been -- that we had on the methodology so we hope that we already satisfied most of the upcoming questions there.

Second section: Shielding analysis.

Since we are using Part 72 equipment under Part 50, we actually looked at both sides and looked at what the applicable dose rate limits are or dose limits are, and we found in general that they are fairly consistent. Part 72 if anything is a little bit more conservative. So we made sure we would cover all of these. Of course the 10 CFR 20 ALARA requirements are active in both Part 50 and Part 72. The next two ones, they are specific to Part 50 and the last two are the dose requirements for Part 72. Our system meets all of this actually by a substantial margin.

Next slide. The thermal analysis. Well, since we will be transporting the fuel in water, there is not really any large thermal challenges. So initially we were looking at what kind of thermal requirements do we really have.

In terms of the fuel, of course, in a reactor, the fuel itself sees a much higher temperature than in any storage.

So for the fuel cladding which is always need to be looked at, we said of course we definitely have to meet the cladding temperature limit of 400 degrees Celsius which is usually used in dry storage, although we are of course are very far away from even getting close to this limit. But we list it here because we need to have some acceptance criteria. For the shielded transfer cask, and the HI-Trac, all components must remain below their temperature limits, which usually is a given, again, based on the fact that we don't really have a very high temperature in our system.

And lastly, although that is not directly linked to the analysis that we performed here, removing spent fuel from one pool to the other and of course from an administrative perspective, the operator in the end has to make sure that the heat load requirements for the spent fuel pool where the fuel is moved to is still satisfied.

Next slide, please. Structural acceptance criteria, we have selected the ASME code subsection ND which is for pressure vessels, for the qualification of both, the shielded transfer cask and the HI-Trac. So each of these two canisters is individually qualified as a pressure vessel so that we meet the stress limits out of these codes. And of course we have various load cases and conditions for normal and accidents. I will talk a little bit more about the accidents later.

We go to the next slide that just briefly summarizes what type of pressures and temperatures we actually looked at. We have set that as the design basis and all the actual values again are well below that.

Next slide, please. Then let me go through the calculations that have been performed and talk about the preliminary results. First, I want to run through the loading requirements.

That is basically what we have set down as assumptions predominantly for the shielding and criticality evaluations. So we have an enrichment limit of 5% which is nothing unusual.

Our burnup upper limit that comes out of the shielding analysis is 55,000 megawatt-days so we are looking at higher burnup of fuel.

That bounds all the fuel that is in the Unit 3 pool and is expected to be coming out of Unit 3. We decided that we set the minimum cooling time to five years, more specifically from the thermal perspective because we want to limit the heat load in the system so we do that by limiting the cooling times for the fuel assemblies.

That's not a big challenge in terms of finding the correct fuel assemblies because

Unit 3 spent fuel pool has a large number of fuel assemblies with fairly long cooling times.

From a criticality perspective, I might further mention 50.68, the regulation that we have chosen that allows credit for burnup of the fuel assemblies. And we essentially are using the burnup, the loading curve and burnup versus enrichment curve, which is the same as already used for the spent fuel pool for the Region 2 spent fuel pool in Unit 3.

So essentially all fuel assemblies that are coming out of the Region 2 spent fuel pool in Unit 3, they would automatically qualify in terms of their burnup versus enrichment curve, for the shielded transfer canister.

And under that condition with that burnup credit, we can load all 12 positions in the basket inside the shielded transfer canister.

However, there are already a small number of fuel assemblies that have not enough burnup to be loaded under these conditions, so we have developed a second loading plan or loading condition where we leave four of the 12 cells empty and only load the 8 cells on the periphery. And under this condition, we essentially can load any fuel assemblies with lower burnups.

The heat generation rate, and that corresponds to a certain burnup and cooling time combination which we have chosen there. We chose a slightly higher heat generation rate for the inner cells and a lower one for the outer cells. We've done that to allow what we call regionalized loading. We want to select fuel assemblies on the periphery which have a slightly lower source term which will reduce the dose rate. So from an ALARA perspective that is beneficial. That is a methodology that has also been implemented under dry storage. We also have that in our dry storage certificates, this option.

Finally, fuel assemblies are required to be intact as we define in the application, and the fuel assemblies can contain inserts because eventually all -- everything out of the Unit 3 pool will be moved -- should be moved into the Unit 2 pool and subsequently into dry storage.

Next slide, please. So for the criticality analyses, I repeat again, Standard Part 50, and preliminary results for what we usually call the normal conditions, soluble -- requires no soluble boron. So it is already the accident with no soluble boron, the maximum Keff is below .94 and for the bounding accident condition, that would be the misloading of a fresh assembly, that would require soluble boron of approximately 700 ppm and again, the maximum K-effective is slightly below .94.

This is both normal and accident condition results applied to both the loading with 12 assemblies with spent fuel or with eight assemblies for fresh fuel. They come out about the same results.

And for the 12 assembly loading, if we go to the next slide, there it is. That is the burnup versus enrichment curve that has to be satisfied. Again, this is exactly the same curve of the Region 2 spent fuel pools in Unit 3 and that also bounds the burnup requirement in the -- I think it's the Region 2-2. They have several regions in Unit 2, of the Region 2-2 in Unit 2. So essentially we've tried to make it as straightforward as possible from an administrative perspective. Once the fuel assemblies are qualified, it's then easy to move them from the Unit 3 pool into the STC, into the Unit 2 pool without further checking the burnup requirement, because they are all consistent with each other.

Next slide, please. Thermal Analyses.

We are using here under Thermal Analyses, the same tools and methodologies that we have been using in our dry storage applications and designs and also in our wet storage, and that's essentially the CFDs, computation of fluid dynamics analysis using the program FLUENT, the 3-D calculations.

We've performed just steady state analysis here, but there are no time limits involved in any of the operations, so the infinite time essentially limit gives us our satisfactory results. And under normal conditions, the fuel cladding temperature is just marginally above 100 degrees Celsius and the bounding accident condition that we've analyzed would be the fire accident outside of the building with the 50-gallon fire; and there the fuel temperature is just only slightly higher because this fire, this is the worst fire condition that can exist, is only fairly short.

The next is the Dose Analyses, and we've done our standard -- that has two parts.

We have done the direct dose analysis, shielding analysis using our standard tool, MCNP, for occupational dose rate and for site boundary dose rates. And we also have looked at -- although we assume and we request that the fuel assemblies are intact, just to demonstrate what kind of effect there could be from potential effluent release, we analyze the effect of effluent release from the casks at the site boundary dose rate.

We used the analysis -- we found that would be the best comparison using the approach that was used in 10 CFR 72 for the effluent release because that was already reviewed and approved in previous versions of our dry storage systems and we assumed some assembly, some failed rods in the assemblies.

So that is essentially just to give a comparison to the direct rates and since the fuel assemblies have a fairly large cooling time, we really see that the dose rates from the effluent release is fairly low.

So the results. Let me list them out here. From an occupational dose rate for a single loading and unloading cycle of the shielded transfer cask, we have compared a conservative estimate of about 1100 mrem.

That compares fairly well to the values that have been calculated and measured for loading and unloading of dry storage casks. This is highly dependent on the operation, in the end.

Our clients, they have a lot of experience with loading and unloading of spent fuel. And it shows, they are over time, they are typically able to reduce the dose rate -- the occupational dose rate substantially by analyzing their steps of operation using long handled tools. There are a lot of things that they can do. This is usually done within the operations and not on our side.

So we find we are quite confident that in the end, the number will be below that. But this is kind of an upper bound of what we would expect.

And I list as the next one I probably should first -- first, mention what the dose rate on the shielded transfer canister is. That is the dose rate for the short period of time when the transfer canister is pulled out of the water and moved over and then put into the HI-Trac. I failed to list that up there.

At that time, there is just the wall of the fuel transfer canister around the fuel assemblies, and there is not any necessity for anybody to be directly next to this canister. So I give first the dose rate at a distance of about 10-meters. We would have about 40 mrem/hour.

And even if you get very close to the system at about a 1-meter distance, you have approximately dose rates of 500 mrem/hour. Usually people would not be that close to the systems but these are dose rates that can easily be managed within a plan, if necessary.

Again, this is only for a very short period of time. And after that, the shielded canister will be placed into the HI-Trac so it gets additional shielding, and then our dose rate drops down to essentially nothing at about 10 meter distance we have dose rates well below 1 mrem/hour. So it's essentially gone.

For the site boundary, under normal conditions we assumed that if transfer takes about eight hours, we've assumed fairly conservative distances initially for these dose rates, only like 50 to 100-meters which is actually much lower than the actual distance to the site boundary. And our direct dose from a single transfer is below 1 mrem.

And the effluent dose is in an order of magnitude below that.

The accident condition that we are considering here is the extended stay of a cask outside for whatever reason, the crawler breaks down or something similar. And we are choosing the 30 day there as a duration which is a value that is consistent with what has been used in Part 72 in dry storage. And then our dose rates are appropriately higher.

Next slide, please: Structural Analyses. For normal conditions we use various types of material and forms of analysis such as ANSYS, and for our accident conditions, we use LS-DYNA, these are all codes that we have been using extensively in wet storage and dry storage analysis. And I didn't want to put any numerical results in here, so I just as a -- as an overview of costs, all stresses are below those that are allowed by the code.

We looked at the seismic stability of the system that is verified under all conditions including in the spent fuel pool and outside of the building. And to verify that the system will actually withstand any -- the limited drop accident that we have, we performed an analysis and showed the G-load deceleration of the system would be below the value that we've determined to be a safe number under Part 72 which is 45-G, so essentially through all this analysis, we show that the system has sufficient structural stability, and structural performance. And this one, just to go back one step, there was one additional question in the white paper, in the letter that we received and that was asking about the drop on to the -- drop of a fuel assembly onto the basket of the STC.

That has also been evaluated and we found that some limited deformation of the basket, but it's so low that it will not affect the performance specifically, the criticality performance of the basket. For those that are familiar with the systems, they will note the wall thickness of the dry storage transportation and storage casks, the wall thickness of the baskets, they are fairly large. Okay.

Next slide. So that was the overview up to now, over the analysis and the brief overview of the results. And the rest of these slides I wanted to just go through the accidents that have been considered. Some of the accidents have been specifically analyzed for this project, some of the accidents that were already analyzed under Part 72 and of course we have taken credit for that. That's the advantage of using Part 72 equipment here. And of course some of the accidents there have already been analyzed under Part 50, so again, these we do not have to repeat them here. They are already covered.

So we will start with an overview of the list of the accidents inside of the Part 50 facility. Of course there is the fuel handling accident, and that is already covered by the fuel handling accident that is evaluated inside the plant.

As a matter of fact, the cooling time of the analyzed fuel handling accident was for a very low cooling time of only 84 hours which would generate a fairly high amount of releasable gases inside the fuel assembly, here we only move fuel assemblies that have already cooled for five years.

There is the fuel assembly misload. There was in the white paper, there was a comment on the number of assemblies that should be considered as misloaded.

We believe that we can implement sufficiently robust controls so that we do not have to consider more than one misloaded assembly from a criticality perspective.

Also there was a question in relation to a thermal misloading of an assembly that had a higher heat load. And we decided that we would implement a temperature monitoring during loading of the basket, so that we exclude -- actually that even while such an assembly might be loaded into the basket, it would never be -- it would be recognized and the loaded system would never be transferred with such a fuel assembly in there.

The accidental drop of a fuel assembly on the top of the STC fuel basket, I've already mentioned that, that is part of the structural analysis. Earthquake seismic stability, we have analyzed that for the shielded transfer cask by itself in the spent fuel pool and for the HI-Trac on the low profile transporter.

Next slide, please. Outside of the Part 50 facility, there is the potential of a slight vertical drop of a loaded HI-Trac 100D and that is only possible during the lifting operation while the HI-Trac is lifted off the low profile transporter into basically the transfer position by the vertical transporter.

Once its actually in that position, it's locked in, so there are -- there is a double failure mechanism which would prevent it from dropping, so this only is possible during a very, very short period of time, but that has been analyzed.

Fire is analyzed. That is the standard 50-gallons of fuel, and it will be ensured that there is during the transfer of the canister between the two units that there is no other combustible material near the system and that is the same approach which has already been taken with the HI-Trac as it has been used now from Unit 2 to the storage facility.

Lightning, these are things that are already evaluated in the HI-STORM FSAR that describes the HI-Trac system. So we just refer to the HI-STORM in that respect. Earthquake, we mention that again here. We've performed the analysis based on the site specific seismic conditions to show that the stability there is demonstrated.

Next slide, please. There is a whole range of environmental loadings, high winds, tornado, tornado-borne missiles. All these have already been analyzed, the HI-Trac has been demonstrated in Part 72 that it withstands all these impacts. The only effect of the tornado-borne missile is that it could lose the water which was used as shielding on the outside, but that is then of course considered actually in the thermal and shielding analysis under the accident condition.

And the last of the events is the extended time and again, that was not only based on the white paper, the letter that was sent, but also based on the assumption that is usually done under 10 CFR 72, the extended time of the fuel transfer canister in the HI-Trac outside of the building. And from a thermal perspective it is covered since all analysis has been used with steady state assumptions and the site boundary dose rates. Those are also assumed, the 30 days there. This is already the next slide.

In the letter, in the white paper, there was also a question in relation to the potential of a tip over of the system. And we have been evaluating that already, and my collegue from Entergy, Joe DeFrancesco, has already pointed out that we will take all the same precautions in the operation and preparation for the operation of the system as we do in Part 72 where we use the same -- essentially the same equipment, namely the vertical cask transporter and the HI-Trac. And we believe based on that, that a tip over of the system, we do not feel that this would have to be considered as a credible event. To just highlight that again, the haul path will be evaluated extensively for all kind of static and dynamic loads. They can be tested and will be tested prior to the transport to make sure that they are stable enough and that a collapse of the haul path will not occur.

Again, we have already pointed out that the pressure, although we have a fairly high overall load, but the pressure due to the size of this vertical cask transporter isn't really very high, and is of the same order of the local pressure of a passenger car. And it would actually would fall for this big structure for the vertical cask transporter for that to tip over. It would require a huge depression or collapse of the roadway for that to occur, and we just -- if you move to the next slide, we've just tried to show that here.

Number one on the left side it shows the vertical cask transporter with the HI-Trac and with the restraints that are built into the transporter to keep the HI-Trac in place. There are actual two restraints on the back and there is a strap around the front so that the HI-Trac actually is immobilized as part of the VCT, of the transporter. And then on the right side, just as an example, it shows a drop of eight-foot.

And it shows that even if the roadway would give way by eight feet, it would create a tilting of the system but it's still far away from tipping over. And so this has been leading us to the conclusion that a tip over of the system from our perspective shouldn't be considered credible and we hope that we can get some feedback in that respect.

And there was the subsequent question in the letter regarding a large release and which from our perspective would be linked to this assumed accident because if there is no tipover of the system, then there is not really any mechanism for any release of any effluent from the system because we have essentially two pressure boundaries, two pressure vessels, one inside the other, and they are both qualified to be structurally capable of remaining intact.

And so, again, this is something that we hope that we can all -- all in the end conclude that these are not credible scenarios.

Although for the release as I mentioned before in my discussion of the dose rates, due to the fact that the fuel has a fairly long cooling time, the concern of a release is actually not fairly high and the dose rates from that would not be expected to be exorbitant. And that concludes my part of the presentation.

MR. WATERS: Thank you Stefan. I am Roger Waters with licensing at Indian Point and I am going to talk about the licensing amendment that we are going to be requesting within the next couple of weeks.

So what we are going to be looking at is a change to the Unit 2 operating license because clearly right now Unit 2 operating license only allows us to possess Unit 2 fuel.

We will put in an amendment request to allow Unit 2 to possess Unit 3 fuel, and we are going to be making amendment requests to the Unit 2 and Unit 3 technical specifications to place controls on the way that the STC is loaded and unloaded within the Unit 2 and Unit 3 spent fuel pools.

Next slide, please. Just a reiteration basically of what I said, for operating license change we are going to be requesting that the operating license be changed for Unit 2 to allow the possession of Unit 3 fuel. And we are going to be more specific than that. We are going to say Unit 3 spent fuel may be transferred to Unit 2 as required.

So that is going to be the operating license changes that are going to be requested.

If you move on to the next slide, this is the Unit 2 technical specifications. So this is going to control the unloading of the STC within the Unit 2 pool. So this is a new technical specification 3.7.15 STC unloading and reloading.

And the LCO is basically, the IP3 fuel must reside within the IP2 racks in its authorized STC cell or be in transit between the two locations. There is no other acceptable locations for that Unit 3 fuel.

The applicability is going to be whenever the STC is in the spent fuel pool with the lid removed. And required actions if the LCO is not met will obviously be to restore compliance with this new LCO.

So what that might be is if we have problems for whatever reason, taking the Unit 3 spent fuel out of the STC and transferring it into the Unit 2 rack, if for some reason we can't put it within its assigned location, then we will put it back within the cell in the STC from whence it came. And then we will verify obviously that we achieved that objective. That's a new tech spec for Unit 2. And associated with that, there's going to be a new request for a new design feature section. And so basically it's going to be a new tech spec, 4.4 for the shielded transfer cask. And it's going to address criticality requirements, drainage, design, and capacity.

And basically under criticality, what we are going to be saying is a reiteration really that the criticality analysis was only done at 5% maximum enrichment and clearly we can't transfer any fuel at a higher enrichment than that. We will place or state limitations on K effective which are based on 10.50.68.

Under drainage, we are going to make a statement that the STC is designed to preclude inadvertent drainage. And under the capacity, we are limiting capacity to a maximum of 12 fuel assemblies.

If we move on to the next slide, it's the Unit 3 technical specifications. We are requesting another new tech spec, 3.7.18, spent fuel assembly transfer.

This is going to control the transfer of the Unit 3 spent fuel that's currently residing in the racks, so it's going to control the movement of that fuel into the STC.

So the LCO is basically going to be 3.7.18; intact fuel assembly shall be classified in accordance with figure 3.7.18-1 and shall have certain attributes. But if we could just skip to the next slide first. This is a new figure that we are going to be putting in the tech spec. And it's analogous to the existing tech spec for loading or placing fuel within the Unit 3 spent fuel racks.

So this is a new figure, burnup versus initial enrichment. The burnout curve is the analytical -- actually the curve that Stefan spoke about earlier, so it's analytically based with actually some margin in it. And we are defining here what Type 2 fuel are and what Type 1 fuel are. Type 2 fuel obviously is the less reactive fuel based on its, you know, region of allowance in this curve. So that's the less reactive fuel, and as such, it can be placed anywhere within the STC.

Type 1 fuel is the more reactive fuel, and it may only be placed around the periphery of the STC. And if you just skip to the next, I'm sorry, if you go forward like two slides, okay, that's barely readable, but it basically shows you the 12 cell locations within the STC. And as I just described, Type 1 and Type 2 fuel is limited to where they can be placed within that STC.

I'm sorry, if you can go back the two slides to the Unit 3 tech specs. I just introduced that the new figure 3.7.18-1 and 18-2. Also there are more limitations and these are also analytically based as you see you are looking at those numbers, you saw those earlier.

Stefan presented those earlier on. These are the analysis numbers that were used in the ALARA shielding calculations and the thermal calculations and the criticality calculations.

All the tech spec values are rooted in the analysis. If we move on to the applicability, that's going to be whenever any fuel assembly is placed in the STC. And action required if we don't meet the LCO, it will be to restore compliance with this new LCO or move fuel to the rack per the existing LCO 3.7.16. So that will ensure that the fuel is placed back into the appropriate location per the existing technical specifications. Then if we move on a couple of slides. These are the Unit 3 tech specs that we are going to be requesting.

These are the new design features and I'm not going to run through all those. Those are exactly analogous to the Unit 2 design features.

So that is basically a summary of what the tech spec and licensing, operating licensing amendments are going to be looking like. Thank you.

MR. CONROY: Okay. And we just have some closing statements that John McCann, our fleet director of licensing would like to provide.

MR. McCANN: I don't have too much to say at this point, but I do want to again thank you for taking the time to meet with us here today.

Clearly we have -- I think we have demonstrated our sense of a need to get this done.

That's why we are asking for it on an expedited schedule. There is no regulatory requirement to have a capability for full core offload, but I'm sure you can appreciate that there is tremendous operating flexibility that goes with that in terms of the way that we can execute an outage, of the risks that are more easily managed with a full core off load which is our preferred way to do refueling outages now for a lot of this work.

Additionally, we can find ourselves in a situation that there are tasks that we need to perform that can't be performed without the capability for a full core off load.

So again, our goal of being here today was to hopefully give you a sense of our readiness to pursue this expeditiously.

We are clearly looking for your help in doing that, understanding that perhaps we are coming in with requests for something that would be a little bit faster than normal. And I think our ability to make that happen will depend an awful lot on our ability to deliver a very high quality product and to promptly and effectively deal with issues whatever issues that come up in the process and my commitment to you is that we will do that.

We really haven't asked you if there are any questions.

MR. BOSKA: Let me say a few words about the question-and-answer period. And I know we have been going a long time and I apologize. I would like to get through the question-and-answer period before we take a break.

So what I would like to do is give the NRC staff first an opportunity to ask questions and then give the public an opportunity to ask questions. And for the NRC staff, I would like to start with staff at the table and then once the table is done, we will call for staff in the audience. I would like to remind NRC staff you will have a further opportunity to ask questions during the proprietary section of the meeting.

So are there any NRC staff at the table that have questions at this time?

MR. CAMPBELL: This is Larry Campbell, Criticality Shielding and Dose Assessment Branch.

I have a few questions and I have some staff sitting right next to me that may have a few specific questions.

But I do -- the couple questions I have, I have looked at your Slide 43, and then I was looking at slides 42 and 26.

Is the intent of Slide 43 to limit enrichment of 5 and 45 gigawatt-days per MTU, or am I looking at this wrong because this seems a little inconsistent with what's on the other slide?

MR. ANTON: The limit of 55,000 is an upper limit for the burnup, and that's been taken from a shielding perspective.

Whereas Slide 43, that defines for certain assemblies a minimum burnup. And that is a burnup that is a minimum requested from a criticality perspective. So essentially if you look at 5%, your maximum burnup is always 55. For fuel type -- for Type 2 fuel, your minimum burnup will be 40,000, and if it's below 40,000, then it would be Type 1 fuel. But the maximum of 55 applies.

MR. CAMPBELL: And then I had a question for the licensee, I guess. The 5 mrem at 1 meter is a high radiation area. Have you considered the possibility of a crane hang-up and what impact that may have on activities in the area, if any at all?

MR. ANTON: Which dose rate are you referring to?

MR. CAMPBELL: On your Slide 30, you indicated that at 1 meter you had 500 mrem.

MR. ANTON: At 1 meter, we could --

MR. CAMPBELL: And the question was, you know, does this affect any operations in the area? Should you have a crane hang-up?

MR. ANTON: Well, the 500 mrem per hour is a higher dose rate, but it doesn't preclude any activities in that area. And of course the procedures which would be put in place there, that would allow for certain activities to be performed there if necessary.

I mean the question with the crane hang-up is -- we are familiar with that question.

MR. CAMPBELL: Okay, we will wait.

MR. BOSKA:. Any questions from NRC staff at the table?

If there are any questions from NRC staff in the audience, could you please come to the speaker at the podium.

MR. WATERS: Mike Waters, Thermal and Containment, Division of Spent Fuel Storage and Transportation.

What is its purpose of the relief valve on the STC?

MR. ANTON: It was in case there would be any over pressurization inside the STC, that the pressure would be released into the HI-Trac overpack, but not any further.

MR. WATERS: Is that part of your accident analysis where you discussed earlier?

MR. ANTON: We have actually not identified any situation where we really would -- that the relief valve would be actuated but we put it in just as a precaution because if there -- if there would be a pressure build-up, it could go into the outer cask and therefore there wouldn't be any further structural damage to the STC.

MR. WATERS: Well, you have an accident dose, can you start me through -- how do you get a release?

What's the accident? How does it get released?

What's the release rates and how do you get the dose?

MR. ANTON: I can explain that.

For the effluent release, we essentially assumed -- if the STC, the seals, they are assumed to be just water tight, not leak tight, as we for example would look in transportation casks. We are not looking for that kind of leak tightness. We just assume that it is bolted down to be water tight. And then further we have assumed that the HI-Trac itself is inefficient to contain any of the effluent and with this condition, we analyzed what the site boundary dose rate would be from the effluent release of the gas in the fuel assemblies.

For the accident condition in there, we -- since we do not consider the tipover as a credible event, we didn't consider a different leakage rate because we are confident that the system will remain essentially water tight under all conditions.

So the accident condition that are presented there is purely the extended stay of the system outside of the building for 30 days as opposed to eight hours.

MR. WATERS: Earlier though you said there was an unlimited transfer times. I'm trying to understand what is the driving mechanism to get the relief, or just the hypothetical relief?

MR. ANTON: It's just the release based on the tightness of the seal, which is the water tightness. And if you -- basically you determine what the activity of the gases inside of the fuel assemblies, we take a certain amount of that is released into the space above -- inside the shielded transfer canister and then from there we use a release rate out into the outside of the cask which is essentially equivalent to what was earlier done in Part 72 with the confinement analysis.

And based on that we've calculated the dose rate.

MR. WATERS: A couple more questions: Are you leak testing the seals before transfer?

MR. ANTON: We will perform a test that verifies that the seals that they are water tight, yes.

MR. WATERS; Last question, why is this being moved wet instead of dry?

MR. ANTON: Well, the point is that it's from an operational perspective. It's much easier and it's also much easier on the fuel.

Number one, you don't have to dry and backfill the system and then flood it with water again. Also the dose rates in the multiple operations that would be necessary for all the drying and backfilling would definitely result in a higher occupational dose rate.

So we tried to make it as easy and as straight forward as possible, ease of operation, basically?

MR. WATERS: Thank you that's all the question I have.

MR. CAMPBELL: I have a follow-up question. The relief valve, so there is no way that that relief valve is going to fail, because I just thought of something. If the relief valve is gone and the water in there could heat up and evaporate, if that's the case, the cladding temperature would go up considerable.

MR. ANTON: You still have the HI-Trac on the outside as the system that holds the whole system in there. So you don't have any overall loss of water from the system.

MR. CAMPBELL: Maybe we just need to see the drawings and configurations.

MR. ANTON: Okay.

MR. McCANN: If I could, I would like to go back to the earlier question, too, maybe to put it in perspective a little bit. I think another number we would like to consider is the fact that although the dose rate is 500 mrem at 1 meter, it's 40 mrem at 30 feet.

If you think of the number of credible situations that a cask and tracker could be in, that could impede operations, there is not too many where you would have to work closer than 30 feet except for perhaps for something on the transporter itself.

MR. BOSKA: Steve, did you have a question?

MR. JONES: Steve Jones, Balance of Plant Branch in the Office of Nuclear Regulation. I just had a couple of questions going back to the project decision, Slide 6.

One was regarding -- I didn't quite understand the member size you were discussing with regard to installing a crane similar to that used in Unit 2.

Is there a greater cantilever length required or does the vertical height in combination with seismic loading result in a greater member size? I'm not quite sure on what is driving that.

PANEL SPEAKER: John Skonieczny can answer that.

MR. SKONIECZNY: Yes. The problem we have run into with putting the Unit 2 type gantry crane in Unit 3, that extra 23 feet in lift height is going to require the member size, the columns themselves to get larger. Right now it barely fits into the Unit 2 fuel storage building. With any larger member sizes, it most likely would not fit in Unit 3, plus we do have other issues with the waste disposal system taking up half the truck bay floor.

MR. JONES: I'm having trouble I guess understanding why additional height adds to member size?

MR. SKONIECZNY: Well now you have higher buckling loads in the members.

MR. JONES: Buckling is the driver, not seismic loading and horizontal --

MR. SKONIECZNY: Well you also have your overturning moments, which is going to drive to much larger ballasts.

MR. JONES: Right. And the second question was, to what extent have you considered a direct transfer to say a smaller --licensing a smaller storage cask that could be handled within the capacity of the Unit 3 superstructure and then moving that directly to dry storage?

MR. McCANN: Let me take a shot at that. Our understanding is that there is really only another cask available that could handle a single fuel assembly at a time.

MR. McCANN: I misunderstood the question.

MR. ANTON: That would essentially require a completely new licensing cycle for a new dry storage system, which definitely would take longer than just creating this link between the capability of the Unit 3 and the existing storage system from that perspective.

Also, of course, in your dry storage system, you want to maximize the capacity, so you want to, for handling and dose rate on the ISFSI. So a dry storage system that would only carry 12 assemblies would be considered fairly inefficient.

NRC STAFF: (inaudible) Spent fuel storage and spent fuel transportation system.

I am looking at Slide 28, essentially the loading curve, for the canister. I guess you mention that this is the same loading curve for Region 2, the spent fuel pool.

I guess that is a little surprising for a 12-assembly canister that you'd reach the same loading curve conclusion. I'm just curious; was there a explicit criticality analysis done of the canister itself?

MR. ANTON: There was an explicit criticality analysis done for the canister. And of course, some conservative assumptions in there and some of them -- there are probably some conservative assumptions in there that may affect the burnup requirement a little bit, but we looked at it and basically said if we can meet the requirement of the spent fuel pool in our canister, then we essentially have showed that the whole system works together.

So we didn't try to optimize that and get more capability out there than necessary. So that might be part of the answer.

NRC STAFF: Okay. And then one more question and I don't know if you can talk about this in the public portion, but can you say anything about the codes that were used for the criticality analysis and the techniques that were used to validate those codes?

PANEL SPEAKER: The codes are the same as we have always used, we use MCMP as the main code. As we do in the wet storage, we have been using CASMO for uncertainty evaluation and the validation of these codes we believe is consistent with the current expectations of the NRR. Let me say that as careful as this.

NRC STAFF: So essentially following the Kopp memo?

MR. ANTON: Yes.

NRC STAFF: Okay. Thanks.

MR. BOSKA: Any other questions from NRC staff?

MR. PARKHILL: Hi, I'm Ron Parkhill with the spent fuel storage and transportation. A few follow-up questions on the relief valve. What is the size of the relief valves, 1x2's, 2x6's?

MR. ANTON: I would have to look that up on the drawing. Do we have that? We probably have that somewhere.

MR. PARKHILL: Are they small, big?

MR. ANTON: It's not just a tiny one. I think it's a sizeable relief valve. But I would have to look up the size.

MR. PARKHILL: And you have two for redundancy or for capacity?

MR. ANTON: I think we only have one relief valve.

MR. PARKHILL: Oh, okay. I guess I have still a little point of confusion here because you designed this transfer cask such that there are no time limits for the transfer operation?

PANEL SPEAKER: That's correct.

MR. PARKHILL: And yet you have relief valves on it. Is that for -- I'm still struggling for why you have relief valves similar as Mike, there. Is it thermal expansion or what is the temperature of the fluid that gets in there, the maximum temperature of the water?

MR. ANTON: The maximum temperature of the water -- I think it's -- let me look it up. It's the design temperature is 150 degrees Celsius and the accident temperature is 200 degrees Celsius.

MR. PARKHILL: So you are over boiling. That's the cladding, isn't it?

ENTERGY: That's the design temperatures.

MR. ANTON: It's significantly lower but with the -- no, we are not boiling water in there, because the system, they are designed to take the internal pressure that results from the expansion and the pressure of the water in there.

MR. PARKHILL: So your steady state heat transfer analysis demonstrates that the water temperature never gets above boiling. So consequently, any discharge that you get from a relief valve is going to be water and not a two phase discharge, then, is that correct?

MR. ANTON: Yes, there will be a release from the gas phase, and --there should in any case not be any boiling in the system because it's -- it withstands the pressure under normal and accident conditions.

MR. BOSKA: Let me just interject something here. If you look at Slide 29. You are saying fuel clad temperature 105 degrees C.

MR. PARKHILL: Yes.

MR. BOSKA: That's at boiling.

MR. ANTON: Not under pressure. That's why the system is designed as a pressure vessel.

MR. BOSKA: So the second temperature on Slide 29 is 116 degrees C?

MR. ANTON: Yes.

MR. BOSKA: Do you know what the saturation pressure is for that temperature?

MR. ANTON: Not off the top of my head, but it's still well below our accident pressure limits are designed at accident pressure limits because for the STC, our accident pressure limit is 65 PSI. And I don't know off the top of my head what the corresponding temperature is, but that is probably --(discussion-inaudible).

MR. PARKHILL: I guess the over all concern is what's driving the need of a relief valve. The code allows you to design a vessel to withstand the pressure and usually you put a relief valve in there if you have some anticipated transient that is going to exceed the structural capacity of it, to protect the vessel.

MR. ANTON: Which we don't and the more I am actual listening here, maybe this relief valve is just creating confusion rather than addressing any --

MR. PARKHILL: Well, it's a nice assurance, I don't want you to necessarily relieve it or get rid of it.

MR. ANTON: We have to definitely make sure that this is -- that there is not any credible way that we actually credit the operation of the relief valve.

MR. McCANN: Maybe it would be worthwhile just to clarify a couple things listening to the questions. You guys correct me if I am wrong here. But the temperature of the fluid will be above boiling temperature at atmospheric pressure. So the canister is designed to be a pressure vessel and I think what you are saying is the relief valve set point and the design pressure of the vessel will be well above the saturation pressure of the fluid for all of the conditions that we anticipate. We would have to demonstrate that.

MR. PARKHILL: So you are saying that the temperature of the fluid will be higher than boiling, 212 at atmospheric pressure?

MR. ANTON: Yes.

MR. PARKHILL: So if the relief valve relieves, what you are saying before is that the fluid would be contained in the outer transporter; does that have a cap on it?

MR. ANTON: Yes.

That is also designed to the same structural requirements so that's also designed as a pressure vessel.

MR. PARKHILL: Okay. That's all. Thanks.

MR. BOSKA: Any other questions from NRC staff?

NRC STAFF: (inaudible) from LAR. One question for the shielding calculation. For your dose rates, did you calculate a dose rate at the dry cask or against the wet cask, meaning with water or without water?

MR. ANTON: They were all calculated with the water.

NRC STAFF: With the water.

So do you consider the situation if you lost the water?

MR. ANTON: No.

NRC STAFF: Would that give you a higher dose rate?

MR. ANTON: Yes, of course. With less water, the dose rate would be higher. But there are two very heavy shielding casks around it.

NRC STAFF: You probably want to consider this kind of evaluation, in case you lose water. During your transportation from your moving, your transporting, is there any (inaudible) or design feature that will prevent you from losing the water?

HOLTEC: There is water inside the canister. Then it's inside another canister that has a lid on the bottom with a seal and another on top with a seal, that is also filled with water.

NRC STAFF: Well, certainly we would be looking into that kind of situation, and your calculation when you submit it to the LAR, or maybe the NRR site will do this evaluation. There is certainly a concern. Okay.

MR. BOSKA: Thank you. Any other NRC staff with questions?

NRC STAFF: I have a couple of questions.

MR. BOSKA: Please use the microphone.

NRC STAFF: Do I need that?

MR. BOSKA: Yes.

MR. TRIPATHI: I'm Bob Tripathi, senior seismic structural engineer. On Slide 33 where you say the stability demonstrated by analysis for HI-Trac and VCD with HI-Trac; and my question is, does your site specific parameters are bounded by your existing analysis for this? Because in the other area, the next slide you say for the tornado-borne missiles and what have you, these are bounded by Part 72 analysis.

MR. ANTON: I believe that the analysis, the seismic analysis have been done with the site specific seismic loads. Whether they are bounded or not by the generic ones, I don't know. They may or may not.

MR. TRIPATHI: And the follow up question, while you said -- have you considered the now infamous newly discovered Ramapo fault which everybody is talking about especially in the northeast area.

There is a newly discovered Ramapo fault. Maybe I'm mispronouncing the actual fault. And it has been in the news lately quite a bit on the East Coast. And Indian Point and a couple of other nuclear power plants are also looking into this. So I was curious whether you looked at it for any implication. The staff has already looked at it independently, but I was wondering if you guys have considered this for Indian Point for this particular operation.

MR. McCANN: I think at this point we are using the current design basis loads.

MR. TRIPATHI: The current committed design basis? Okay. Thank you.

MR. BOSKA: Any other NRC staff with questions? All right. Are there any members of the public that have questions, if you would please come up to the microphone.

All right. We did have some call-ins from members of the public. And these questions are directed to the NRC staff. But Entergy is allowed to answer them if they so desire.

The first question is, how many assemblies in the Unit 3 spent fuel pool have failed fuel rods? And I would say from an NRC point of view, that we do not know the answer to that. I don't know if Entergy wishes to answer, you're not required to answer.

MR. GUMBLE: Floyd Gumble, reactor engineering supervisor. There are approximately a dozen fuel assemblies in the Unit 3 pool that have failed rods. I do not have the exact number here but it is a comparatively small number given the capacity of the pool at 1345 assemblies.

MR. BOSKA: Thank you. The next question was, since there are assemblies with failed fuel rods, what special methods will you use to transfer these fuel assemblies from the Unit 3 spent fuel pool?

MR. GUMBLE: We were not planning to use this mechanism for failed fuel at all.

MR. BOSKA: All right. So you would be planning to use something like a transportation cask that's qualified for failed fuel?

MR. GUMBLE: That is correct. And that is to be handled farther in the future.

MR. BOSKA: All right, thank you.

The next question is, given the fact that the fuel assemblies are being handled more frequently, there appears to be an increase in risk here, and will the NRC consider additional controls for the licensee to follow for the handling operations, like work-hour limitations, limitations on weather conditions, et cetera?

I think I would say in the area of work hour limitations that we have recently implemented a new regulation on work hours, and that we consider that regulation satisfactory to cover this situation along with all the other situations at the nuclear plants.

There would be a question on weather condition limitations or other controls.

Anybody?

MR. GUMBLE: I can answer that, if I might. We have administrative controls at the plant based on severe weather, for example, blizzard. One of the things we do if severe weather is anticipated or comes upon us in the case of a tornado, is that fuel handling is secured.

MR. BOSKA: All right. Thank you. Any other questions from members of the public?

All right. We have reached the end of the question-and-answer period.

We will take a 20-minute break, and let me just say that at this point we are going to shift into closed session for proprietary information, so I do need to request that all members of the public leave the room at this time.

So we will start again at 3:00. Thank you.

(Whereupon the meeting was adjourned)

Enclosure 3

NRC Handout

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

NRC Meeting With Entergy Indian Point Units 2 & 3 Spent Fuel Transfer

Nuclear Regulatory Commission- Headquarters

Introductions

Nuclear Regulatory Commission

- Robert Nelson, Deputy Director, DORL, NRR
- Ray Lorson, Deputy Director, DSFST, NMSS
- John Boska, Project Manager, DORL, NRR
- John Goshen, Project Manager, DSFST, NMSS

Entergy

- John McCann, Director-Licensing
- Patric Conroy, Director-Nuclear Safety Assurance
- Ed Weinkam, Senior Licensing Manager
- Robert Walpole, Manager-Licensing, Indian Point

Purpose of Today's Meeting

- This is a Category 1 meeting. The main purpose is to allow the licensee to discuss their proposal with NRC staff.
- The public is permitted to observe the meeting, and ask questions of NRC staff during the public question and answer period.

Purpose of Today's Meeting

- Entergy will present their proposal to transfer spent fuel from Indian Point Unit 3 to Indian Point Unit 2 for further transfer to the independent spent fuel storage installation which already exists at the site
- NRC technical reviewers will have the opportunity to ask questions in order to understand the scope of Entergy's proposal
- Stakeholders will have an opportunity to address questions to the NRC staff regarding the proposal
- The second half of the meeting will be closed to the public due to discussion of proprietary information

Agenda

- Introduction
- Overview of Spent Fuel Transfer Project - Entergy
- Overview of Licensing Actions - Entergy
- Questions by NRC Staff
- Questions by Members of the Public (call 301-415-2123)
- Meeting Closed for Proprietary Discussion

Enclosure 4

Licensee Handout

NRC Pre-submittal Briefing on License Amendment Request for Indian Point Energy Center Inter-Unit Transfer of Spent Nuclear Fuel

June 15, 2009

Introductions

- Entergy - Indian Point Energy Center
- Holtec International

Objectives

- Present Plans to Transfer IP3 Spent Fuel to the IP2 Spent Fuel Pool and from there into Dry Cask Storage at the Existing ISFSI

Agenda

- Project Overview
- Review Fuel Transfer Equipment Design
- Fuel Transfer Operation Process
- License Acceptance Criteria and Results
- Licensing Actions Requested
- Closing
- Q&A
- Design and Operations (proprietary discussion)

Terms and Definitions

- FSB refers to Indian Point Fuel Storage Building Unit 2 or Unit 3
- STC is the Shielded Transfer Canister
- HI-TRAC 100D is the licensed transfer cask part of the HI-STORM 100 Cask System Certificate of Compliance 72-1014
- VCT is the Vertical Cask Transporter
- LPT is a Low Profile Transporter that uses Hilman rollers or Air Pads
- LCO is technical specification Limiting Condition for Operation

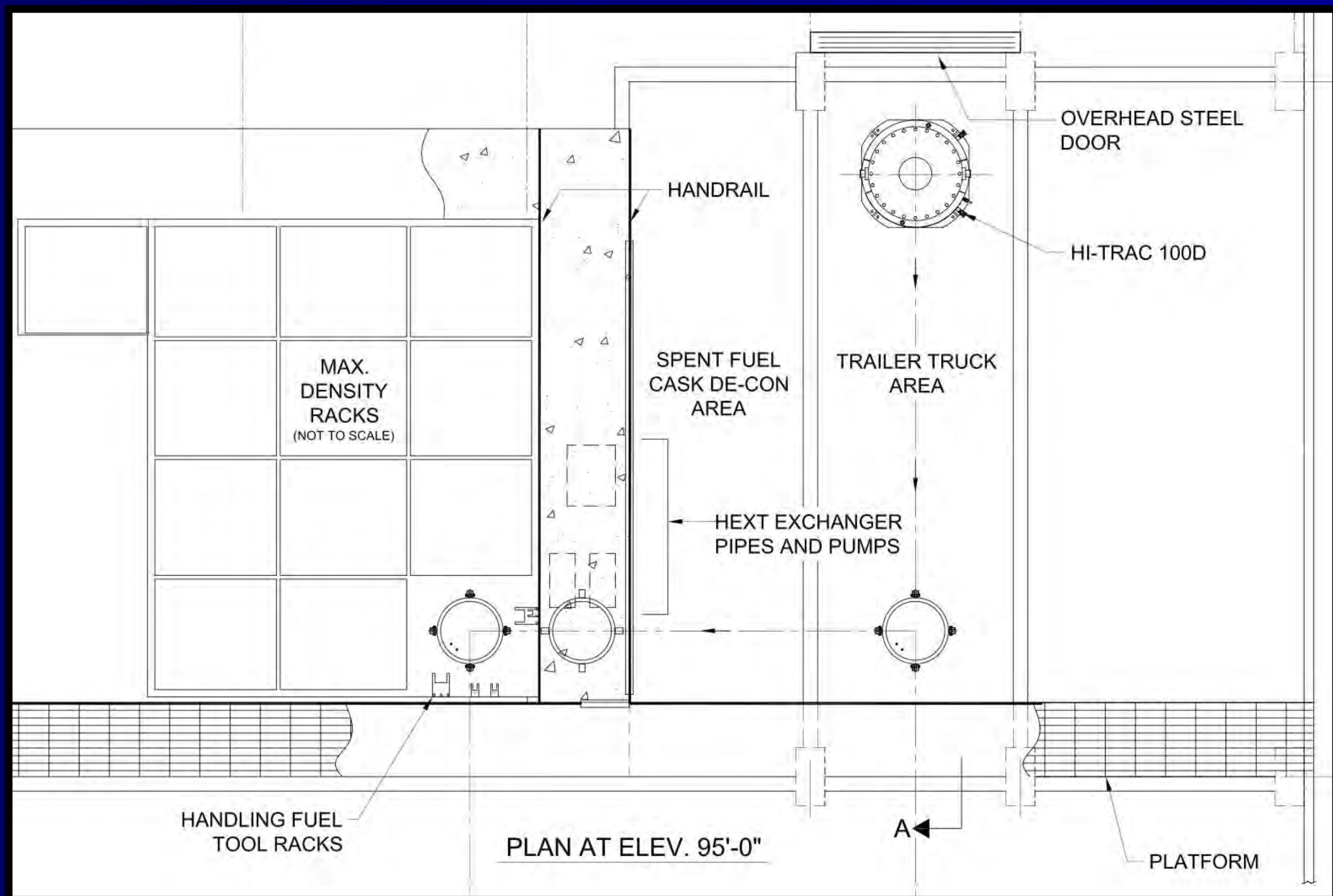
Project Decision

- Fuel Inventory
  - Unit 2 Full Core Offload maintained by Dry Fuel Storage
  - Unit 3 Lost Full Core Offload Capability
- Background
  - Crane Capacity Upgrade from 40 to 100-tons not feasible
  - Unit 2 type Gantry Crane not feasible for Unit 3
    - Greater lift height, approximately 23’ more
    - Existing plant system interferences
- Inter-Unit Fuel Transfer
- Transfer from Unit 2 to Dry Cask Storage

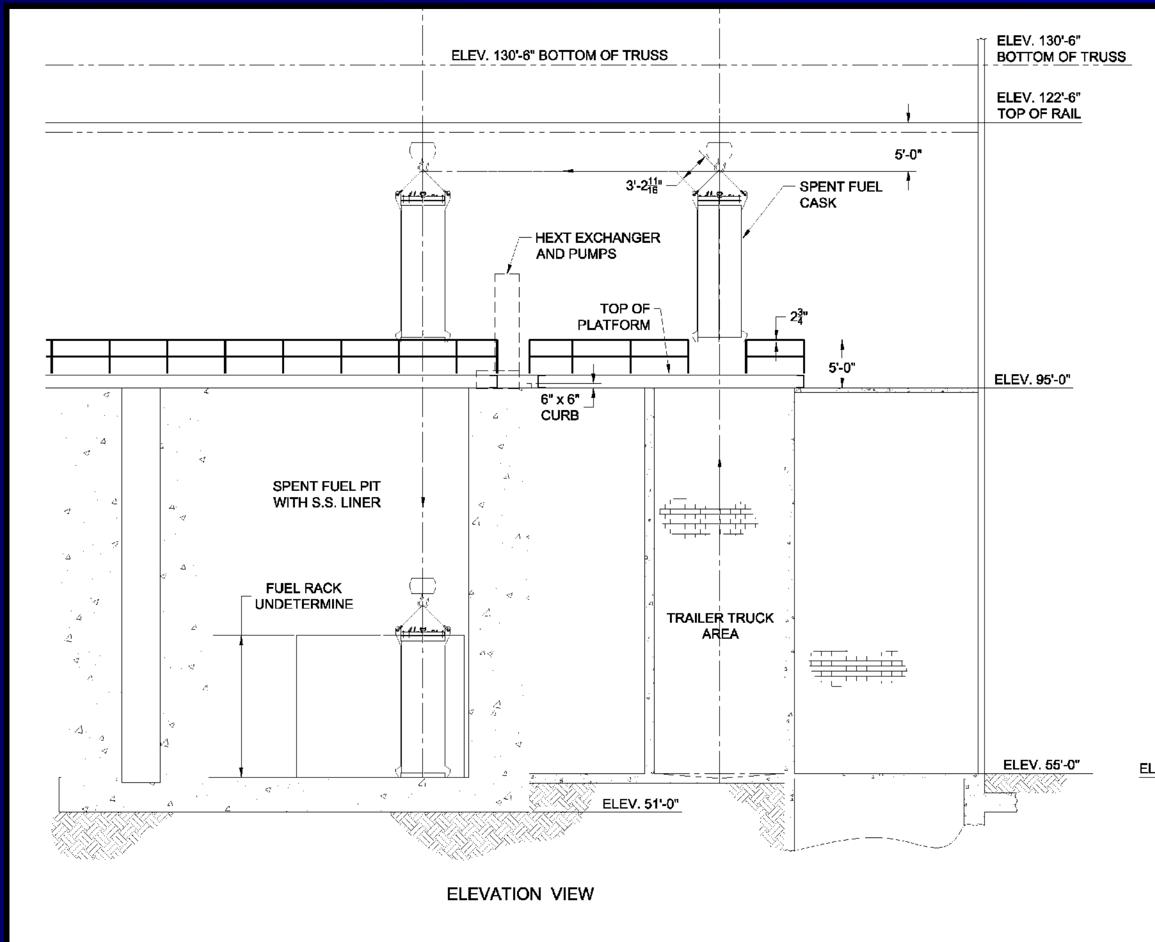
Project Scope

- 40-ton Crane Replacement – NUREG 0554
- Unit 3 FSB Truck Bay Door Replacement and Interference Relocations
- Unit 3 FSB Truck Bay Floor Leveling
- Roadway Haul Path Improvements
- Protected Area Boundary Modification
- Fuel Transfer Equipment
- Transfer Unit 3 Fuel Assemblies to Unit 2

Unit 3 FSB Layout



Unit 3 FSB Elevation



Fuel Transfer Equipment

- New Shielded Transfer Canister
- Existing HI-TRAC with Solid Top Lid
- Special Lifting Devices – ANSI N14.6
- Low Profile Transporters
- Existing Vertical Cask Transporter

STC Design

- Approximate Dimensions - 52” diameter x 183” high
- Lifting attachments - Trunnions for Lift Yoke
- Loaded STC Handling weight:
  < 40-tons with rigging
- STC fuel capacity - 12 fuel assemblies
- Fuel basket design - Stainless Steel with Metamic®

STC Design (cont)

- Lid closure design - bolted flat head with elastomer seal
- Safety features - relief valves (RV)
- Maintenance requirements - calibration RVs, Trunnion inspections, general inspections
- Codes and standards
  - ASME B&PV Code Section III, Subsection ND
  - ANSI N14.6

HI-TRAC Transfer Cask Design

- Using Unit 2 existing HI-TRAC 100D
- Solid Top Lid
- Lid closure bolting with elastomer seal
- Maintenance requirements - per Part 72 FSAR and general inspection
- Codes and standards
  - ASME B&PV Code Section III, Subsection ND
  - ANSI N14.6

Fuel Transfer Operation

- Move STC into Unit 3 spent fuel pool
- Fuel loading, lid installation
- Remove STC from pool and place in HI-TRAC
- Remove water from STC for air space and install lid bolting
- Using LPT, Move outside Unit 3 FSB
- HI-TRAC moved with VCT
- Move HI-TRAC/STC into Unit 2 FSB with existing LPT
- Using Unit 2 Gantry Crane lift STC and place in spent fuel pool cask handling area
- Remove STC Lid and move fuel to Unit 2 racks

Transport Haul Path

Licensing Actions

- Licensing Amendment Request Contents:
  - Unit 2 Operating License
  - Unit 2 New Technical Specifications
  - Unit 3 New Technical Specifications
- Submittal Date: June 2009
- Requested Approval Date: Aug 2010

Acceptance Criteria

- Criticality
- Shielding
- Thermal
- Structural

Criticality (50.68)

- The effective neutron multiplication factor (keff) shall be less than 0.95 with the STC fully loaded with fuel of the highest anticipated reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity.
- Reactivity effects of abnormal and accident conditions shall be evaluated to assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95, with credit for soluble boron to offset the accident condition.

Criticality (cont)

- 10CFR50.68, “Criticality Accident Requirements”
- 10CFR50, Appendix A, General Design Criterion 62, “Prevention of Criticality in Fuel Storage and Handling.”
- NUREG-0800, Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling”

Criticality (cont)

- R.G. 1.13, “Spent Fuel Storage Facility Design Basis”
- ANSI ANS-8.17, “Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors”
- GL-78-11, “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications”
- L. Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants”

Shielding

- 10 CFR 20.1101(b) ALARA operations exposure will be as low as reasonably achievable.
- 40 CFR 190.10 fuel cycle normal operation ≤25 mrem whole body, ≤75 mrem thyroid
- 10 CFR 100 accident condition ≤25 rem to whole body and ≤300 rem to the thyroid from iodine exposure
- 10 CFR 72.104 ≤25 mrem whole body, ≤75 mrem thyroid, ≤25 mrem other critical organs
- 10 CFR 72.106 ≤5 rem deep-dose equivalent

Thermal

- The spent fuel cladding temperatures must be below 400 ºC in accordance with SFST ISG-11 Rev. 3
- The STC and the HI-TRAC components must remain below their component temperature limits as specified in the HI-STORM 100 FSAR for the HI-TRAC
- Total heat load transferred to the Unit 2 fuel pool will be limited by the design basis heat load specified in the Unit 2 FSAR

Structural

- STC is designed to meet the Level A stress limits of the ASME B&PV Code, Section III, Subsection ND
- HI-TRAC inner shell and pool lid evaluated and top lid designed for the effects of internal pressure stress limits of ASME B&PV Code, Section III Subsection ND
- Load Cases for Normal and Accident Conditions

Design Basis

Criterion	STC	HI-TRAC
Design Pressure, psig	50	30
Accident Pressure, psig	65	40
Design Temperature, ºC	150	150
Accident Temperature, ºC	200	200

Preliminary Results

- STC Contents
- Criticality
- Shielding
- Thermal
- Structural

STC Loading Requirements

- Initial enrichment ≤ 5 wt% U235
- Burnup ≤ 55,000 MWD/MTU
- Cooling time ≥ 5 years
- Minimum burnup as a function of the initial enrichment when all 12 assemblies are loaded
- No minimum burnup when assemblies are loaded only in the 8 outer cells, with 4 inner cells empty
- Heat generation rate ≤ 1,105 Watt for inner cells, ≤650 Watt for outer cells, ≤9 kW total
- Fuel assemblies must be intact
- Fuel assemblies may or may not contain inserts

STC Criticality Analyses

- Standard Part 50 criticality analysis
- Preliminary Results
  - Normal Conditions
    - Soluble Boron: 0 ppm
    - Maximum keff: < 0.9400
  - Bounding Accident Condition (Misloading of fresh assembly)
    - Soluble Boron: 700 ppm
    - Maximum keff: < 0.9400
  - Burnup as a function of initial enrichment (12 assemblies) see next slide

Burnup vs Enrichment

Nominal Initial Enrichment (wt% ^{235}U)	Minimum Burnup (GWD/MTU)
1.8	0
2.0	5.50
2.5	12.75
3.0	18.20
3.5	23.90
4.0	29.75
4.5	35.00
4.95	40.00

Thermal Analyses

- 3-D Steady State Computational Fluid Dynamics Analysis (FLUENT)
  - no time limits for transfer
- Preliminary Results
  - Normal Conditions
    - Fuel Cladding Temperature 105 ^{o}C
  - Accident Condition (Fire)
    - Fuel Cladding Temperature 116 ^{o}C

Dose Analyses

- Standard Shielding Analysis (MCNP) for Direct Dose (occupational and site boundary)
- Standard 10CFR72 effluent release analysis, assuming 1 failed assembly
- Results
  - Occupational dose (conservative estimate): 1100 mrem
  - HI-TRAC Dose Rate at 10 m 0.03 mrem/hour
  - Site Boundary, normal conditions, per transfer (8 hours)
    - Direct Dose: 0.6 mrem
    - Assumed Effluent: 0.01 mrem
  - Site Boundary, accident conditions (30 days)
    - Direct Dose: 65 mrem
    - Assumed Effluent: 0.8 mrem

Structural Analyses

- Strength-of-material and Finite Element Analysis (ANSYS) calculations for normal conditions
- Transient non-linear finite element analyses (LS-DYNA) for vertical drop accident condition
- Preliminary Results
  - All stresses below those allowed by code
  - Seismic stability during transfer is verified
  - g-load under drop accident below 10CFR72 acceptance limit

Credible Accidents and Off-Normal Events

- In Part 50 Facility
  - – Existing Fuel Handling Accident
  - – Fuel Assembly Mis-load
  - – Accidental drop of a fuel assembly on the STC fuel basket
  - – Earthquake; Seismic Stability demonstrated by analysis for STC in pool and HI-TRAC on LPT

Credible Accidents and Off-Normal Events (cont)

- Outside Part 50 Facility
  - Accidental vertical drop of loaded HI-TRAC 100D (during lifting operation)
  - Fire; spillage and ignition of 50 gallons of combustible fuel, 4.8 min duration, 1475 ºF, max ambient 100ºF
  - Lightning; evaluated in HI-STORM FSAR
  - Earthquake; Stability demonstrated by analysis for HI-TRAC and VCT with HI-TRAC

Credible Accidents and Off-Normal Events (cont)

- Outside Part 50 Facility (cont)
  - – Environmental Loadings; extreme environmental phenomena, such as high winds, tornado, and tornado-borne missiles, Reg. Guide 1.76, ANSI 57.9, and ASCE 7-88, are considered in the certification of HI-TRAC 100D in Docket No. 72-1014 and bound the conditions at Indian Point.

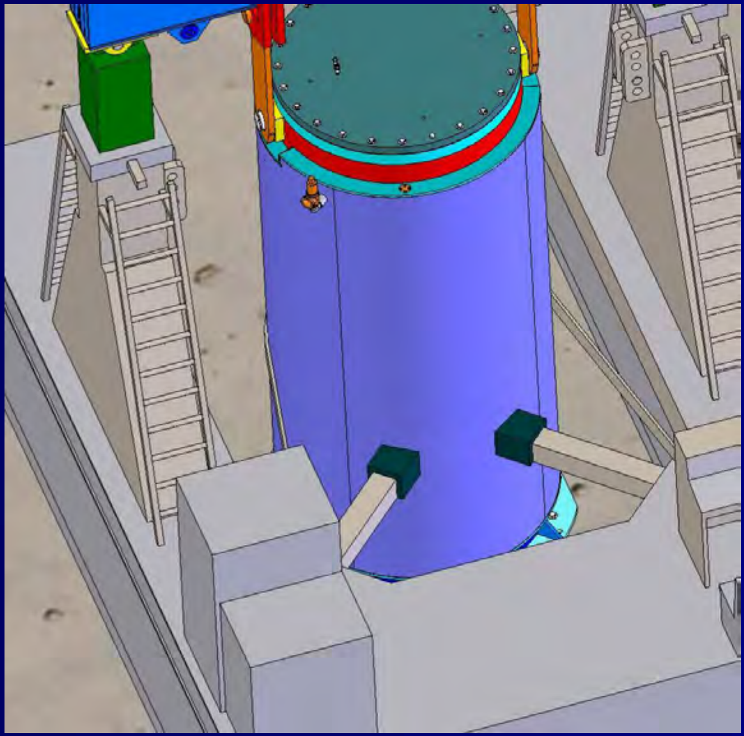
Credible Accidents and Off-Normal Events (cont)

- Outside Part 50 Facility (cont)
  - Extended time of STC residence in the HI-TRAC: thermal hydraulic analyses steady state; site boundary dose calculations assume 30 days

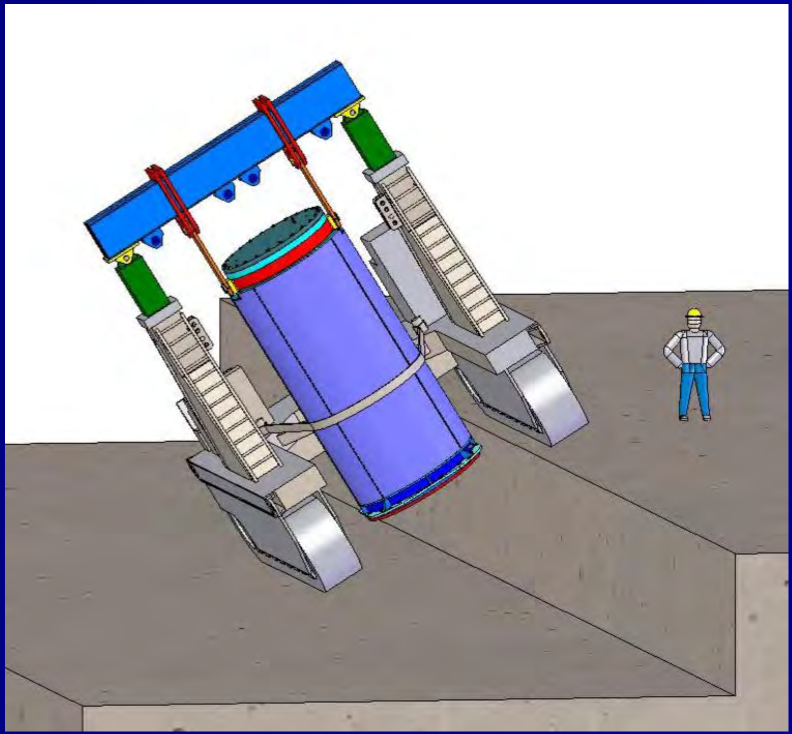
VCT Tip-over Resistance

- Haul path will be evaluated for static and dynamic loads
- Haul path can be tested to 150% of the load prior to transport
- Transporter with fully loaded HI-STORM
  - 56 psi ground pressure (180-ton overpack)
  - 35 psi for a typical passenger car
- Greater than 10 ft depression for Tip-over

VCT Tip-over Resistance



HI-TRAC Restraint



Hypothetical 8 ft depression

Licensing Actions

- Licensing Amendment Request Contents:
  - Unit 2 Operating License
  - Unit 2 New Technical Specifications
  - Unit 3 New Technical Specifications

Unit 2 Operating License

- Possess special nuclear material produced from operation of Units 2 and Unit 3
- Unit 3 spent fuel may be transferred to Unit 2 as required

Unit 2 Technical Specifications

- New TS 3.7.15 STC Unloading and Reloading
- LCO 3.7.15 IP3 fuel must reside in IP2 racks, in its authorized STC cell, or be in transit between the two locations
- Applicability: Whenever the STC is in the spent fuel pool with lid removed
- Actions: Fuel assembly not in required location, restore compliance with LCO 3.7.15
- SR: Verify fuel location by administrative means

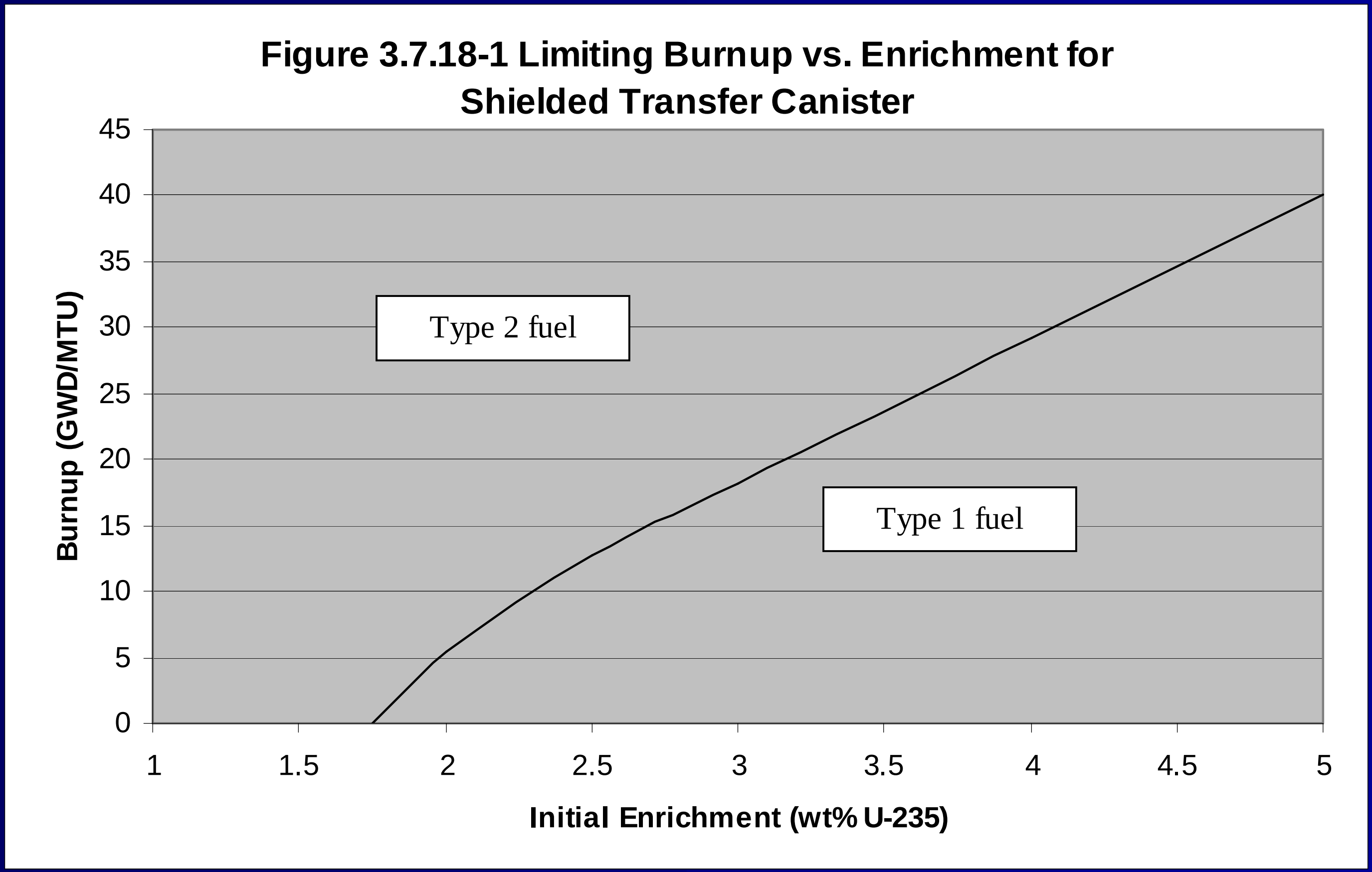
Unit 2 Technical Specifications

- New Design Features
  – New TS 4.4 Shielded Transfer Canister
  – TS 4.4.1 Criticality
  – TS 4.4.2 Drainage
  – TS 4.4.3 Capacity

Unit 3 Technical Specifications

- New TS 3.7.18 Spent Fuel Assembly Transfer
- LCO 3.7.18 Intact fuel assemblies shall be classified in accordance with Figure 3.7.18-1 and shall have:
  - Post irradiation cooling time ≥5 years
  - Assembly burnup ≤ 55,000 MWD/MTU
  - Decay heat ≤ 650 watts (any STC cell)
  - Decay heat ≤ 1105 watts (inner four STC cells)
  - Type 1 fuel can only be loaded into peripheral cells and the inner cells must be empty
  - Type 2 fuel can be loaded in any cell
  - Cell locations are defined in Figure 3.7.18-2
- Applicability: Whenever any fuel assembly is placed in the STC
- Actions: Does not meet LCO, restore compliance with LCO 3.7.18 or move fuel to rack per existing LCO 3.7.16
- SR: Verify by administrative means

Unit 3 Technical Specifications



Unit 3 Technical Specifications

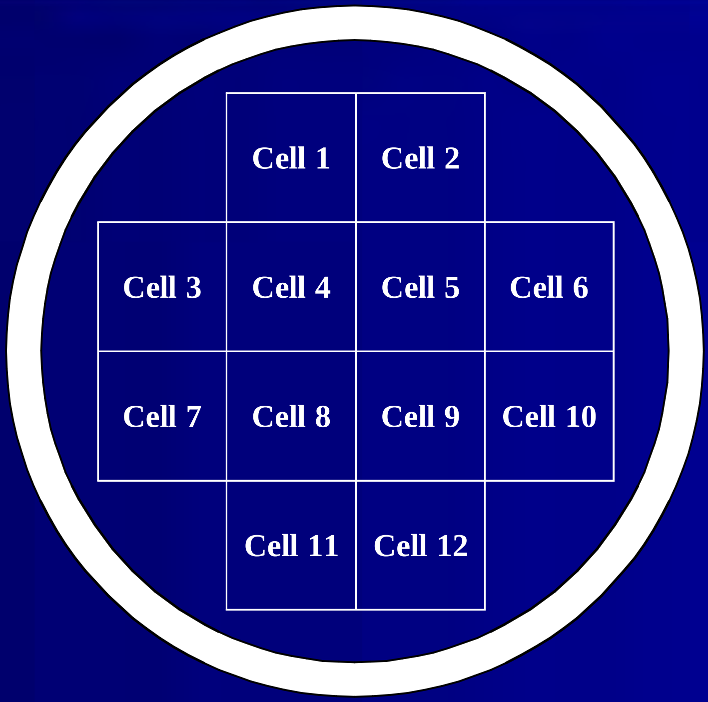


Figure 3.7.18-2: Shielded Transfer Canister Layout
(Top View)

Unit 3 Technical Specifications

- New Design Features
  - – New TS 4.4 Shielded Transfer Canister
  - – TS 4.4.1 Criticality
  - – TS 4.4.2 Drainage
  - – TS 4.4.3 Capacity

Closing Statements

Questions?

July 6, 2009

LICENSEE: Entergy Nuclear Operations, Inc.

FACILITY: Indian Point Nuclear Generating Unit Nos. 2 and 3

SUBJECT: SUMMARY OF JUNE 15, 2009, MEETING WITH ENTERGY ON THE TRANSFER OF SPENT FUEL FROM UNIT 3 TO UNIT 2 AT INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 (TAC NOS. ME0408 AND ME0409)

On June 15, 2009, a Category 1 public meeting was held between the Nuclear Regulatory Commission (NRC) and representatives of Entergy Nuclear Operations, Inc. (licensee) and Holtec International, Inc. at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the licensee's plan to transfer spent fuel from the spent fuel pool at Indian Point Nuclear Generating Unit No. 3 (IP3) to the spent fuel pool at Indian Point Nuclear Generating Unit No. 2 (IP2), and from there to the independent spent fuel storage installation which already exists at the site.

The licensee presented information on the method they would use to transfer the spent fuel, using a specially designed canister and an existing shielding cask to move the spent fuel across the site within the protected area of the site. The licensee discussed their plan to submit license amendments for IP2 and IP3 to request NRC approval for this fuel transfer. A transcript of the meeting, excluding the discussion of proprietary information, is provided as Enclosure 1. A list of attendees is provided as Enclosure 2, but may not be all-inclusive. The NRC's handout is provided as Enclosure 3, and the licensee's handout as Enclosure 4.

No Public Meeting Feedback forms have been received. Please direct any inquiries to me at 301-415-2901, or by email to John.Boska@nrc.gov.

/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286
Enclosures:
1. List of Attendees
2. Transcript
3. NRC Handout
4. Licensee Handout

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OFFICE	LPL1-1/PM	LPL1-1/LA	LPL1-1/BC
NAME	JBoska	SLittle	NSalgado
DATE	7/1/09	7/6/09	7/6/09

OFFICIAL RECORD COPY

DATED: July 6, 2009

SUMMARY OF JUNE 15, 2009, MEETING WITH ENTERGY ON THE TRANSFER OF SPENT FUEL FROM UNIT 3 TO UNIT 2 AT INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 (TAC NOS. ME0408 AND ME0409)

PUBLIC
LPL1-1 R/F
RidsNrrDorlLpl1-1
RidsNrrPMJBoska
RidsNrrLASLittle
MGray, RI
RidsOGCRp
RidsAcrsAcnw&mMailCenter
RidsNrrOd
RidsRgn1MailCenter
JGoshen, NMSS
EBenner, NMSS
LCampbell, NMSS
MWaters, NMSS
ABarto, NMSS
DTang, NMSS
DPickett, NRR
SBurnell, OPA
RNelson, NRR
KWood, NRR
ZLi, NMSS
BTripathi
GThomas, NRR
KRopon, NMSS
JParillo, NRR
EThompson, NMSS
LCruz, NMSS
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