Canadian Nuclear Safety Commission	**Commission canadienne de sûreté nucléaire**
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March 27th, 2014	**Le 27 mars 2014**
Public Hearing Room 14th floor 280 Slater Street Ottawa, Ontario	Salle des audiences publiques 14e étage 280, rue Slater Ottawa (Ontario)
Commission Members present	**Commissaires présents**
Dr. Michael Binder Mr. Dan Tolgyesi Dr. Sandy McEwan Ms Rumina Velshi Mr. André Harvey	M. Michael Binder M. Dan Tolgyesi Dr Sandy McEwan Mme Rumina Velshi M. André Harvey
Secretary:	**Secrétaire:**
Mr. Marc Leblanc	M. Marc Leblanc
General Counsel:	**Avocate générale :**
Ms Lisa Thiele	Me Lisa Thiele

TABLE OF CONTENTS

TABLE OF CONTENTS

Ottawa, Ontario

--- Upon commencing on Thursday, March 27, 2014 at 9:08 a.m. / L'audience débute le jeudi 27 mars 2014 à 9 h 08

CMD 14-M11

Opening remarks

M. MARC LEBLANC: Bonjour, Mesdames et Messieurs. Bienvenue à la réunion publique de la Commission canadienne de sûreté nucléaire.

We have simultaneous translation. Please keep the pace of speech relatively slow so that the translators have a chance to keep up.

Des appareils de traduction sont disponibles à la réception. La version française est au poste 2 and the English version is on Channel 1.

Please identify yourself before speaking so that the transcripts are as complete and clear as possible.

La transcription sera disponible sur le site Web de la Commission dès la semaine prochaine.

I would also like to note that this proceeding is being video webcast live and that archives of the proceedings will be available on our website for a three-month period after the closure of the proceedings.

I would also ask you to please silence your cell phones and other electronic devices.

Monsieur Binder, président et premier dirigeant de la CCSN, va présider la réunion publique d'aujourd'hui.

President Binder...?

LE PRÉSIDENT : Merci, Marc.

Good morning and welcome to the meeting of the Canadian Nuclear Safety Commission. Mon nom est Michael Binder, je suis le président de la Commission canadienne de sûreté nucléaire et je vous souhaite la bienvenue. And welcome to all of you joining us via webcast.

I would like to start by introducing the Members of the Commission who are here with us today.

On my right is Monsieur Dan Tolgyesi. On my left is Dr. Sandy McEwan, Ms Rumina Velshi and Monsieur André Harvey.

We already heard from our Secretary Marc Leblanc and we also have with us Ms Lisa Thiele, General Counsel to the Commission.

MR. MARC LEBLANC: The *Nuclear Safety and Control Act* authorizes the Commission to hold meetings for the conduct of its business. Please refer to the updated agenda that was published on March 20th for the complete list of items to be presented today.

In addition to the written documents reviewed by the Commission for today's meeting, CNSC staff will have an opportunity to make presentations and Commission Members will be afforded an opportunity to ask questions on the items before us.

Mr. President...?

CMD 14-M12.A

Adoption of Agenda

THE PRESIDENT: Okay. With this information, I would like to call for the Adoption of the Agenda by the Commission Members as outlined in Commission Member Document CMD 14-M12.A.

Do we have concurrence?

For the record, the Agenda is adopted.

CMD 14-M13

Approval of Minutes of Commission Meeting held February 5 and 6, 2014

THE PRESIDENT: I would like to move on to the approval of the Minutes of the Commission meeting held on February 5th and 6th, 2014. The minutes are outlined in Commission Member Document CMD 14-M13.

Any comments, additions, deletions?

I hear no such comments, no changes. Therefore, I'd like to get approval for the minutes.

Okay. So for the record, the minutes are approved.

I'd like to proceed now to the Status Report on Power Reactors, which is outlined in CMD 14-M14, and Dr. Rzentkowski, I understand you're going to make the presentation. Please proceed.

CMD 14-M14

Status Report on Power Reactors

DR. RZENTKOWSKI: Thank you very much and good morning, Mr. President and Members of the Commission.

I have no further updates for today on the Status Report on Power Reactors presented as CMD 14-M14.

However, I would like to mention that since our last status report presented to the Commission on February 6th, 2014, there have been neither safety significant events nor events of particular public interest.

I would also to bring to your attention an update regarding Bruce B, Unit 5 leak from the Heat Transport System instrument line which was reported at the Commission Meeting held on August 21st, 2013. This is Section 1.2 of the CMD.

The leak resulted in the spill of approximately 400 kilograms of heavy water. Bruce Power staff promptly isolated the leak, monitored contamination and initiated actions to identify, contain and clean up the spill. As reported, the event had no impact on the safety of the workers and the environment.

CNSC staff was tasked with providing the Commission with an update on the Root Cause Analysis and implementation of corrective measures as required. CSNC staff has received and reviewed the additional report on Root Cause Analysis and concluded that Bruce Power corrective measures implemented as a consequence of this event are sufficient to prevent this type of event in the future.

Mr. Ken Lafrenière, Regulatory Program Director for the Bruce Power site, will describe the measures in a moment.

No future update to the Commission on this event is planned.

Thank you very much.

Ken...?

MR. LAFRENIÈRE: Thank you, Greg.

CNSC staff received the report, the preliminary report, on June 23rd. That was followed by a detailed report on August 2nd of 2013. The additional reports were received on August 9th of 2013. And the final additional report was received on January 15th of 2014 and that additional report contained the Root Cause Analysis.

As Mr. Rzentkowski mentioned, staff have reviewed the root cause and were satisfied with the measures taken by Bruce Power for this event. The test in question was repeated successfully in the units and the test is no longer quarantined.

Some of the root causes were essentially that the test failed to identify critical steps that the operator needed to perform and the test has been since revised and repeated.

That completes the update. Thank you.

THE PRESIDENT: Have you finished?

DR. RZENTKOWSKI: Yes. This concludes the Status Report on Power Reactors. Thank you.

THE PRESIDENT: Okay. Thank you.

I'd like to open up the floor for questions. We'll start with Monsieur Harvey.

MEMBRE HARVEY : Merci, Monsieur le Président.

J'aurais une question à propos de Gentilly-2.

On dit que :

* La vidange du circuit du caloporteur est présentement en cours selon les procédures revues et acceptées par le personnel de la CCSN. +

Et d'un autre côté, on dit que :

* Tous les canaux du réacteur ont été drainés à l'aide de la machine à chargement du combustible. +

C'est deux choses différentes, ça, la vidange du circuit du caloporteur. Pouvez-vous élaborer un peu sur ça, dire quelle est la... Moi, j'avais l'impression qu'une fois que c'était vide, que tout était drainé, mais ce n'est pas le cas.

Dr RZENTKOWSKI : Oui. J'aimerais maintenant céder la parole à M. Alex Leblanc, superviseur du bureau site de la centrale Gentilly-2, par téléconférence.

M. A. LEBLANC : Bonjour. Je m'appelle Alex Leblanc.

Pour répondre à la question de monsieur Harvey, c'est que le drainage du caloporteur, la façon que le réacteur est placé est qu'il est plus bas que les générateurs de vapeur, et alors, quand ils vident le caloporteur, ils drainent certaines sections, et quand on arrive aux tubes de force du réacteur, on doit visiter chacun des tubes de force avec la machine à chargement. Alors, il y a 380 canaux, et chacun des canaux doit être visité individuellement pour les vider du caloporteur.

MEMBRE HARVEY : Mais est-ce que, à l'heure actuelle, tout est terminé?

M. A. LEBLANC : Présentement, les tubes du réacteur sont vidés, et ce qu'il reste à faire, c'est vider certaines petites sections de caloporteur, principalement des lignes d'instrumentation.

MEMBRE HARVEY : Ah, je m'excuse.

M. A. LEBLANC : Je peux peut-être élaborer là. Ça fait que la façon... Il y a beaucoup de sous-systèmes dans le caloporteur. Alors, quand on draine le caloporteur, il y a certaines sections... C'est drainé par sections, et les grosses sections, les sections contenant le plus grand volume de caloporteur, ces sections-là ont été drainées.

Et ce qui reste présentement à faire, c'est vider certaines des sections qui contiennent beaucoup moins de volume de caloporteur. Mais, en général, les plus grands volumes, comme je l'ai expliqué, sont déjà drainés. Et les plus petites sections, il faut qu'elles soient visitées individuellement, ce qui rallonge beaucoup le processus de drainage de caloporteur.

MEMBRE HARVEY : Non, ça va. Là, je comprends bien le système.

Puis quelle va être l'étape suivante une fois que ça va être terminé?

M. A. LEBLANC : O.K. Bien, présentement, Hydro-Québec, là, ce qu'ils ont fait, c'est que le caloporteur a été placé dans quatre grands réservoirs, et, par la suite, le caloporteur va être mis en barils.

MEMBRE HARVEY : Et tout ça va être entreposé sur le site, c'est ça?

M. A. LEBLANC : Je crois qu'il y a des plans présentement pour vendre le caloporteur à une autre centrale nucléaire.

MEMBRE HARVEY : Bon, merci.

I have another question for --

LE PRÉSIDENT : Est-ce que je pourrais poser une question?

Quand est-ce qu'on va recevoir le plan de démantèlement?

M. A. LEBLANC : Excusez, je n'ai pas compris la question.

LE PRÉSIDENT : Quand est-ce qu'on va recevoir le plan de démantèlement de Gentilly-2?

M. A. LEBLANC : Le plan de démantèlement de Gentilly-2?

LE PRÉSIDENT : Oui. C'est ça.

M. A. LEBLANC : Je n'ai pas la réponse.

Présentement, ce qu'on a reçu de Gentilly-2, c'est le plan de fin d'exploitation. Ce plan-là explique comment mettre la centrale... ou les étapes qu'Hydro-Québec vont entreprendre pour mettre la centrale à l'état de stabilisation, qui devrait se produire là en 2015, puis, par la suite, qu'est-ce qu'ils vont faire pour maintenir certains systèmes de la centrale en fonction, par exemple, qu'est-ce qu'ils vont faire avec le combustible irradié.

Et, pour le moment, c'est prévu de

démanteler la centrale en... je crois que c'est 2055. Ça fait qu'au niveau du plan de démantèlement, je ne peux pas vous dire là quand est-ce qu'on va le recevoir. Je crois qu'il y a des discussions encore avec la gestion d'Hydro-Québec.

MEMBRE HARVEY : Il n'y a pas eu de décision de prise encore pour voir si Hydro-Québec allait aller avec un démantèlement rapide ou une plus longue période de temps?

M. A. LEBLANC : Pas à ma connaissance. Je crois que ça, c'est une décision encore sous évaluation chez Hydro-Québec, et puis, bien nous... en tout cas, moi, je ne suis pas impliqué dans ces discussions-là. Il y a possiblement des gens dans la haute gestion de la CCSN qui en connaissent plus sur ce dossier-là que moi.

Mais à ce que je sache, en ce moment, le plan, c'est de démanteler la centrale à plus long terme là, autour de 2055 là, dans ce coin-là.

LE PRÉSIDENT : O.K. Monsieur Leblanc, je pense que monsieur Rinfret a la réponse. Alors, vas-y.

M. RINFRET : O.K. François Rinfret, directeur pour Darlington.

Je pourrais simplement rajouter que présentement le titulaire a soumis son plan de déclassement préliminaire, et le plan préliminaire, c'est une condition du permis. Ça, c'est disponible.

La centrale n'a pas encore déposé ses orientations ou sa stratégie à plus long terme. Il n'y a pas d'obligation.

L'objectif présentement sur lequel toute la division travaille pour Gentilly puis le titulaire aussi, c'est de s'assurer d'arriver à l'état de stockage sûr, qui va prendre encore quelques mois. Pour le reste, les plans ne sont pas encore présentés.

LE PRÉSIDENT : O.K. Merci.

Monsieur Harvey...?

MEMBER HARVEY: My last question was about Pickering Unit 8, about the --

THE PRESIDENT: Sorry, I should have recognized that we have Mr. Gilbert from OPG who is with us, I hope, online, willing to answer some questions.

Mr. Gilbert...?

MS SWAMI: Laurie Swami for the record, Vice-President Nuclear Services for OPG.

Mr. Gilbert won't be joining us this morning but I'm available to answer any questions you may have.

THE PRESIDENT: Okay. Thank you.

MEMBER HARVEY: Okay. I wanted to know what -- it seems to be very simple here, it has been fixed and everything is okay, but what is the importance to maintain the temperature below 13.1 and what could be the impact of having a higher temperature for a long period of time?

You can answer, the staff maybe.

DR. RZENTKOWSKI: I will ask Mr. Miguel Santini to respond to this question. He's responsible for the regulatory oversight of the Pickering site.

MR. SANTINI: Miguel Santini for the record.

The incremental temperature of the plant, being A or B, are numbers available between the Ministry of Environment of Ontario and the operator of the plant, in this case OPG. I'm not familiar with how they come up with those numbers. All I know is that OPG is in the process of initiating discussions with the Minister to revise those numbers. Perhaps OPG could have more details on that.

MS SWAMI: Laurie Swami for the record.

When you look at the Delta T or the temperature change across the facility, you consider the impact on the environment from that change in temperature, and so the studies that are done look at what the potential impact would be to fish and other aquatic effects. So that's the importance of maintaining the Delta T.

THE PRESIDENT: Can you give us a little bit about what was the cause for the malfunction here?

MS SWAMI: Laurie Swami.

We had some issues with electrical equipment in our screenhouse which required that we shut down our condenser cooling water pumps, one pump, and when we do that the temperature change across the plant increases as a result of ongoing operation. We fixed that situation and returned our condenser cooling water pump to service.

THE PRESIDENT: Thank you.

MEMBER HARVEY: My point was just to -- if there's a problem and it's impossible to lower the temperature for a certain period of time, what would Ontario do about that? They would close the plant? What is the measure that could be taken by Ontario if it continues to be like this for a long period of time?

MS SWAMI: Laurie Swami for the record.

We don't intend to operate with an exceedance of this particular value. It was a failure that caused this situation to occur. If we had to operate in that mode, we would obviously have to take additional measures to ensure that we met the requirements of our Environmental Compliance Approval Certificate. So we would do that. It may mean changing the power rating if that was an opportunity, but it certainly is not something that we would plan for because it's our intent to operate in our current design configuration.

MEMBER HARVEY: Was it the first time you encountered this type of problem?

MS SWAMI: Laurie Swami for the record.

We from time to time have had situations where our Delta T or our temperature across our plant has exceeded this particular value. As we would normally do, we investigate every time that that occurs. We put in place corrective actions to prevent it from recurring.

So the process is the same as with any area where we don't meet a particular environmental compliance requirement, we look at the cause, we investigate, we put in place the corrective actions to prevent recurrence.

MEMBER HARVEY: Does Darlington have the same requirement about the 13.1?

MS SWAMI: Laurie Swami, for the record. The Darlington design is different from the Pickering design and it has specific features to manage the temperature change in the lake than does Pickering. Pickering has a surface water intake and discharge, whereas Darlington has a deep water intake and a fairly long discharge channel, if you will, with specific design features to manage the Delta T. So it's a very different design and, as a result, different requirements.

MEMBER HARVEY: Thank you.

THE PRESIDENT: Thank you. Ms Velshi...?

MEMBER VELSHI: So just picking up on this discussion on this Certificate of Approval Limit Exceedance, when you shut down the condenser cooling water pump, then you would have known that you would have exceeded that then?

MS SWAMI: Laurie Swami, for the record. Yes, we would know that it was a potential to exceed that and so would have been part of the recovery from that event.

MEMBER VELSHI: And what are the consequences of exceeding that? Is there a penalty, is there a fine that gets imposed?

MS SWAMI: Laurie Swami, for the record. That would be something that the Ministry of the Environment would consider as they looked at the performance of the plant overall.

We would, of course, have to report that to the Ministry of Environment and they would consider whether to impose fines or not.

MEMBER VELSHI: And you said this happens from time to time. Is that like a couple of times a year or dozens of times a year?

MS SWAMI: Laurie Swami, for the record. It's not a few times a year, it's more infrequent than that. It does occur occasionally and that would be over the extended period of time that the plant is in operation.

MEMBER VELSHI: Thank you. I have a question on the Bruce Unit 5 heat transport leak. I may not have followed what was reported around what were the key root causes of that. I thought I heard that there may have been some procedural inadequacies.

Can you please elaborate on that? I'm assuming this test happens on a regular basis, so would they have run into similar issues in the past using this procedure?

DR. RZENTKOWSKI: Mr. Ken Lafrenière will elaborate on this point, but I would also like to mention that Frank Saunders, the Vice President of Bruce Power, is present in the room as well and may provide more details.

MEMBER VELSHI: Thank you.

MR. LAFRENIÈRE: Thank you, Greg. Ken Lafrenière, for the record. So the root cause of the event was in fact the test had not identified critical steps.

CNSC staff reviewed the original test and the revised test and found that it should prevent reoccurrence of the event. The test they were doing is not a frequently performed evolution, as a matter of fact it's done perhaps I think on an annual basis, so they did not have a lot of experience doing that test.

The contributing causes also of the event was a little bit of training -- inconsistencies in the operator training, basically on how much to tighten the valve, physical resistance things. The operator was worried that over tightening the valve would cause damage to the circuit, so he did not tighten the valve sufficiently essentially. That was a contributing cause, they have since addressed that with their -- in their operator training programs.

Another contributing cause was, since the test was done very infrequently, on an annual basis, the two -- the pre-job briefing and the peer review stages of the process had individuals involved that had actually never performed that test. So another fix that Bruce Power has instituted is to require for infrequently performed evolutions to have an experienced person who has performed a test conduct a pre-job briefing so that the more -- the critical evolutions in the steps, they would have more experience behind that discussion prior to the test taking place.

Mr. Saunders might want to add a bit more to our review.

MR. SAUNDERS: Yes, perhaps just to take a little step back and explain what this test is. It is a flow transmitter, it is on the emergency core injection system, so it is a system that doesn't actually operate ever actually, so we need to confirm that the flow transmitter is accurate, so it's an annual test that actually confirms that the flow of transmitter measures accurate flow in the ECI.

To do this, that transmitter is normally looking at heat transport system pressure, so to do this we valve it out of the heat transport system for a short period and we valve in a test loop. The test loop is at considerably lower pressure than the heat transport system, so you are depending on the two valves that isolate it from the heat transport system that keep the two pressures separate.

The system was perhaps a little poorly designed in that, in most cases when we do these kind of things we actually provide a gauge or some means that you can actually confirm that the isolation has occurred. This system didn't actually have any clear indication, so one of the problems was we have a system which you don't use that often and which doesn't have a clear indication.

The other issue was the clarity around the operating procedures as Mr. Lafrenière mentioned. These days on our procedures we spend a lot more time identifying critical steps and issues that might be of concern.

This is an older procedure, it didn't have some of those items in it, so one of the main corrective actions here was to rewrite the procedure to very clearly identify the critical steps such as isolating the heat transport system and we did actually look and find some ways that we can verify that the system -- that it is properly depressurized before we valve in the test loop.

On the longer term we are also changing the operator basic skills training program to account for, you know, the odd system like this which doesn't have as easy a way of identifying. One of the key things in the operator of process is to teach them to verify de-energization, whether you are talking electrical or pressure, you want to verify before you work on a system that the system is in fact de-energized.

Some systems that's a little more difficult than others, so going back to basic operator skills to make sure that this particular kind of problem is understood and introduced there.

THE PRESIDENT: You just mentioned that you do it annually.

MR. SAUNDERS: Yes.

THE PRESIDENT: Is that the first time this kind of mistake has been done?

MR. SAUNDERS: Yes. In this system that's the first time we have had a problem with it, a combination of circumstances, but as you see a couple of weaknesses there and over time the opportunity came about for something to go a bit wrong; so correct the weaknesses to make sure it doesn't happen again.

And we are of course doing extensive condition, looking to see if there are other kind of infrequently performed operations which have similar risks in them that we can identify in advance.

THE PRESIDENT: Thank you. Mr. Tolgyesi...?

MEMBER TOLGYESI: Merci, Monsieur le Président. I have just a few questions to Pickering about this water temperature.

Is the value of Delta T different from summer to winter time?

MS SWAMI: Laurie Swami. The temperatures are managed across the plant given the current conditions. Generally speaking they are the same between winter and summer in the various seasons.

There are some times when there are other impacts at the plant that are considered in our environmental compliance approval that allows us to have different values, but they are on a specific reasons for those that are mentioned in the approvals we have.

MEMBER TOLGYESI: Here you are talking about average temperature. What does that mean, that it varies from where to where?

MS SWAMI: Laurie Swami, for the record. So generally when the plants are in operation there is, generally speaking, about a 10-degree across the plant on each unit, so on average it would be about there, depending on the temperature of the water coming into the facility. And so that Delta T generally is in that range of 10 to 11. It doesn't vary that significantly unless units are shut down.

MEMBER TOLGYESI: And when you were talking about this electrical problem with the intake screening wash, does it mean that the screen washing is on a continuous basis, or it is fragmented so many times or so many hours per day or per period?

MS SWAMI: Laurie Swami, for the record. The screen wash system is available regularly for the incoming water, so it is a system that we use to ensure there is no debris coming into the facility which would harm our equipment. So it operates on a regular basis.

MR. SANTINI: If I may, to help to understand -- my name is Miguel Santini, for the record. As you may recall, I believe last month or prior to that we made -- we also reported an exceedance and it was because of the ice build-up.

So basically the occurrences of these exceedances happen when there are equipment problems such as these or when there are major problems where they have to shut down some of the pumps in order to deal with the ice build-up in the Bay.

On some occasions -- and we also reported that -- they even had to shut down the unit because of the ice build-up.

MEMBER TOLGYESI: And my last question, Mr. President. You were saying that the nature of additional measures, one you were mentioning is power rating could change.

Are there some other measures you could use before you reach power rating lowering?

MS SWAMI: Laurie Swami, for the record. There are limited other opportunities for us to effect the Delta T across the facility.

Some of our plants -- so, for instance, Pickering 5 to 8, we have a system that would allow us to add tempering water to the outfall which would adjust the outfall temperature and thereby change the Delta T across that facility. So there are opportunities like that.

Pickering A doesn't have that same opportunity. So, as an example, we could perhaps operate additional CCW pumps if a unit was shut down during that period, something of that nature, but there are very few opportunities to do more than manage the Delta T as necessary.

MEMBER TOLGYESI: Just for personal information, what's the temperature of the lake in the summer and in the winter?

MS SWAMI: Laurie Swami for the record.

The temperature of the lake varies very significantly. As Mr. Santini explained, we had ice blockage. The temperature of the lake is essentially 0 degrees Celsius, and it can range up to high 20s, if not higher, in the summer months. So there's quite a variety of temperatures that we manage as water coming into the plant.

THE PRESIDENT: Thank you.

Dr. McEwan...?

MEMBER McEWAN: No questions.

THE PRESIDENT: Thank you.

I note that we have no event initial report this time around. Are there any other events that CNSC staff would like to bring to the attention of the Commission at this time?

DR. RZENTKOWSKI: With respect to the nuclear power plants, no. This concludes our update. But I believe we have one more stemming from an event in Halifax.

M. RÉGIMBALD : Bonjour, Monsieur le Président et Membres de la Commission. Je m'appelle André Régimbald, et je suis responsable de la réglementation des substances nucléaires.

Je voudrais vous faire part ce matin... Ce n'est pas l'incident de Halifax. Ça va être un peu plus tard, Greg. C'est un autre.

Nous n'avons pas présenté de rapport par écrit. Alors, j'y vais a capella, comme on dit.

L'incident s'est produit la semaine dernière, plus précisément le jeudi 20 mars 2014, à la mine de fer de la compagnie Cliffs Québec Mine de Fer Limitée, à Fermont au Québec, au cours duquel une dizaine de travailleurs, qui ne sont pas des travailleurs du secteur nucléaire, auraient vraisemblablement reçu une dose de rayonnement supérieure à la limite de dose pour le public de 1 mSv.

La compagnie Cliffs est titulaire d'un permis de la CCSN l'autorisant à utiliser des jauges nucléaires contenant du Cs-137 pour contrôler le débit de minerais qui circule sur des convoyeurs d'alimentation un peu partout dans l'usine de la mine.

L'incident est survenu lorsque les travailleurs s'affairaient à des travaux d'entretien autour des convoyeurs quand ils se sont aperçus que deux des quatre jauges nucléaires installées dans la zone de travail étaient encore en position ouverte. C'est-à-dire que l'obturateur de ces deux jauges n'avait pas été cadenassé en position fermée avant le début des travaux selon la procédure. Les deux autres jauges situées dans la même zone étaient quant à elles en position fermée.

L'information que nous avons à ce moment-ci indique que les travailleurs ont bel et bien reçu une dose de rayonnement supérieure à 1 mSv. Par contre, il n'y a pas suffisamment de détails pour déterminer avec précision la dose exacte que chaque travailleur a reçue, mais tout indique que les travailleurs ont reçu une dose inférieure à 50 mSv. Pour mettre le tout en perspective, 50 mSv est, comme vous le savez, la limite de dose pour les travailleurs du secteur nucléaire.

Les doses aux travailleurs impliqués dans l'incident n'ont eu aucun effet néfaste sur leur santé, et les travailleurs ont été rassurés par la compagnie que leur santé n'est nullement compromise.

Lorsque l'incident est survenu, la compagnie a immédiatement fait stopper les travaux et a informé la CCSN dès le lendemain, soit le 21 mars.

En fin d'après-midi le 21 mars, la CCSN a émis un ordre à la compagnie l'obligeant à fournir des renseignements précis qui serviront au calcul des doses, tout en interdisant la compagnie à effectuer tout autre travail d'entretien dans des endroits où sont situées des jauges nucléaires, ainsi que l'interdiction de faire le montage ou le démontage de jauges nucléaires, jusqu'à ce que des mesures correctives jugées satisfaisantes par la CCSN auront été mises en place par la compagnie pour prévenir une répétition de ce genre d'incident.

Le 22 mars, la compagnie a fourni

à la CCSN un rapport détaillé de l'incident avec les renseignements demandés et a indiqué qu'elle poursuivait son enquête.

Le personnel de la CCSN est en train d'examiner l'information fournie par la compagnie afin de déterminer avec précision les doses reçues par les travailleurs et s'assurera que la compagnie effectuera un suivi approprié avec les travailleurs afin de les tenir au courant des doses reçues une fois que celles-ci auront été confirmées.

Alors, ceci est un rapport très préliminaire à ce moment-ci. Nous retournerons devant vous au moment voulu avec une présentation plus complète lorsque les causes et les circonstances exactes de l'incident auront été éclaircies, que les doses aux travailleurs auront été confirmées et que le titulaire de permis aura fourni un plan d'action pour mettre en oeuvre les mesures correctives jugées satisfaisantes.

Merci beaucoup.

LE PRÉSIDENT : Merci beaucoup.

Des questions? Monsieur Harvey...?

MEMBRE HARVEY : La première

question, c'est comment on peut dire qu'il n'y a pas eu d'effet sur la santé si on n'a pas une idée des doses qui ont été reçues?

M. RÉGIMBALD : Je vais demander à mes collègues de la Division de radioprotection de fournir un peu plus d'information à ce sujet. Merci.

M. B. THÉRIAULT : Donc, c'est Bertrand Thériault ici, spécialiste en dosimétrie.

Donc, les estimés préliminaires de dose indiquent que ça serait sous le cap du 50 mSv, probablement de l'ordre de dose maximale de quelques dizaines de mSv. Étant donné que les effets néfastes pour la santé sont observés au-dessus de 100 mSv ou plus, donc, c'est pour cette raison qu'on dit, à ce point-ci, qu'il ne semble vraiment pas y avoir d'effets à long terme.

MEMBRE HARVEY : Mais qu'est-ce qui vous amène à dire que ça été quelques dizaines de mSv? L'information, vous la prenez où?

MR. B. THÉRIAULT : Alors, dans le rapport préliminaire qu'on a reçu de Cliffs Quebec Iron Mining, alors, un spécialiste en radioprotection s'est rendu sur les lieux puis a pris des mesures avec un radiamètre pour estimer le débit de dose, le nombre de mSv par heure.

Puis avec l'information fournie par les registres d'entrée en espace clos et ainsi que des entrevues avec les travailleurs, à savoir pour obtenir un estimé de la période de temps passée par chaque travailleur à proximité de la source, il a été possible d'avoir des estimés préliminaires.

Ce que ce rapport nous indique, c'est que ces estimés seraient conservateurs, seraient prudents. Mais, évidemment, comme monsieur Régimbald l'a indiqué, on attend le rapport final complet pour avoir une idée plus claire des doses.

MEMBRE HARVEY : Il y avait quand même plusieurs travailleurs. C'était six ou huit travailleurs qui étaient... Mais ces travailleurs-là n'étaient pas tous au même endroit non plus, j'imagine, et les doses peuvent être différentes dépendamment de l'endroit où le travailleur était placé?

M. B. THÉRIAULT : Oui, effectivement. Alors, dans certains cas, certains travailleurs ont passé très peu de temps, d'autres plus de temps.

Le nombre de travailleurs selon ce rapport préliminaire indiquerait un total de 25 travailleurs en tout qui auraient été impliqués, avec 16 d'entre eux qui pourraient potentiellement avoir excédé la limite de 1 mSv, mais encore, c'est à confirmer.

MEMBRE HARVEY : Pour revenir dans un autre ordre d'idée, sur le rapport préliminaire, est-ce qu'il y avait une explication que deux de ces jauges-là étaient cadenassées puis les autres ne l'étaient pas était en fonction... Comment une chose comme ça peut arriver si... Est-ce que c'est parce que les jauges ne fonctionnent pas toutes en même temps ou est-ce que c'est un oubli?

M. RÉGIMBALD : André Régimbald ici.

C'est exactement ce que nous essayons de comprendre. L'information préliminaire qu'on a reçue indique que les travailleurs auraient omis de cadenasser deux des jauges en position fermée. Alors, l'enquête va déterminer les causes, pourquoi c'est arrivé, et aussi si c'est une erreur humaine ou une défaillance mécanique.

Alors, nous attendons, et c'est pourquoi nous reviendrons plus tard avec la présentation plus complète pour vous expliquer exactement les causes et les circonstances.

MEMBRE HARVEY : Mais est-ce que vous avez, dans les dossiers de la Commission, le nom des personnes qui sont susceptibles d'utiliser ces jauges-là? Est-ce que ces gens-là étaient qualifiés et est-ce que, s'ils étaient qualifiés, ils étaient inscrits dans vos registres?

M. RÉGIMBALD : Les travailleurs du titulaire de permis ont reçu la formation nécessaire et ont des procédures à suivre. Maintenant, comme je vous l'ai expliqué, la procédure exige que les jauges soient cadenassées en position fermée avant que des travailleurs puissent effectuer des travaux d'entretien.

Alors, l'information qui sera fournie au cours de l'enquête permettra d'éclaircir les raisons exactes pourquoi deux jauges n'ont pas été cadenassées en position fermée.

MEMBRE HARVEY : Mais mon point visait le travailleur qui s'occupait des jauges. Je me disais, est-ce que ce travailleur-là était qualifié pour utiliser les jauges? Puis s'il l'était, est-ce que, lorsque quelqu'un est qualifié, il est inscrit dans un registre de la Commission?

M. RÉGIMBALD : Le travailleur qui procédait au contrôle des jauges était... selon le rapport, était l'électricien qui s'occupe des travaux d'entretien.

Mais il y a aussi un responsable de la radioprotection qui est responsable de veiller à ce que les travaux dans des endroits où sont installées des jauges nucléaires sont effectués selon la procédure qui exige le cadenassage en position fermée.

MEMBRE HARVEY : Vous parlez d'un électricien, mais cet électricien-là, est-ce qu'il a besoin d'être qualifié aussi pour s'occuper des jauges?

M. RÉGIMBALD : Oui. Selon les documents du titulaire de permis, l'électricien doit avoir la formation nécessaire pour effectuer la procédure exigeant la fermeture des jauges.

MEMBRE HARVEY : O.K. Merci.

M. RÉGIMBALD : Et nous reviendrons avec toute l'information.

LE PRÉSIDENT : O.K. On a assez de questions sur une situation qu'on ne sait rien.

MEMBRE TOLGYESI : Vous avez dit que la personne qui s'occupe de cadenassage doit être formée. Est-ce que c'est une formation qui est donnée dans l'entreprise, par l'entreprise, ou nous exigeons comme Commission qu'il doit suivre une formation...

LE PRÉSIDENT : Quand on va recevoir les détails, on va savoir tout. Maintenant, c'est assez. On a discuté... Je pense qu'on a assez de renseignements et c'est un rapport compréhensif pour le moment.

Aimeriez-vous répondre à cette question?

M. RÉGIMBALD : Lorsque le demandeur de permis nous remet la demande de permis, il doit fournir les programmes de formation pour les employés. Alors, nous allons examiner les programmes de formation et nous allons étudier les circonstances de l'incident pour voir s'il y a eu des manquements à cet égard.

LE PRÉSIDENT : O.K. Merci beaucoup.

Is there any other event that you

would like to report on?

Go ahead.

MR. RINKER: My name is Mike Rinker, I am the Director of the Nuclear Processing Facilities Division at the CNSC.

If I could, I would like to update the Commission on the status of the Shield Source Incorporated facility located in Peterborough.

Just for some background, the facility has been operating more than 25 years as a tritium processing facility and about two years ago the Commission, on its own motion, restricted the activities of this operator to no longer process tritium and last year of March, 2013, Shield Source Incorporated informed the CNSC that they would no longer want to continue operating at this location.

So they commenced some cleanup activities, some decontamination activities at their facility under their operating licence and the CNSC staff would like -- we are pleased to inform you that these activities are complete.

CNSC lab personnel and inspectors have been on site entirely during the cleanup activities and can confirm through our own monitoring results and verification of Shield Source Incorporated monitoring results that the facility is clean and fit for a new tenant, a non-nuclear, non-regulated tenant to move in.

And the state of the environment is much improved, as we would have expected since the facility has been not processing tritium for two years. Tritium -- although the tritium levels were safe two years ago, they are much lower now and so as of yesterday CNSC staff have submitted a CMD to the Commission to request authorization to release the facility from licensing through the issuance of an abandonment licence.

THE PRESIDENT: Thank you. Questions? Ms Velshi...?

MEMBER VELSHI: A couple of questions. So you said the tritium levels are lower, the facility is clean. Does that mean they are still higher than background levels?

MR. RINKER: Mike Rinker, for the record. So background would refer to the environment. They are higher than background levels.

If I could give some examples, though, the tritium in air is near zero, about maybe a factor of 100 to 1000 times lower than what they were during operations and they were safe during operations, but you can measure.

Tritium in soil are -- it is measurable, but it is very low and we are really talking about a strip of soil along the edge of the facility and there are no credible scenarios that would enable anyone to get exposed to unsafe levels of tritium in soil.

If somebody were to excavate that soil and handle it manually for the course of a week, the dose consequences are in the order of a microsievert or two.

And, finally, groundwater, they are measurable. There are a number of groundwater wells there, all except one are below the drinking water standard. So they are not drinking water sources.

This facility is built on a landfill, it will never be a drinking water source, but the tritium levels are below the drinking water standard, with the exception of one that was at 15,000 becquerels per litre, has since decreased to about 8,000 becquerels per litre over the last two years and the drinking water standard is 7,000.

We are going to continue to monitor that well to make sure that it continues to decrease.

MEMBER VELSHI: So as far as ongoing monitoring, other than that one well, will there be any other ongoing monitoring of that facility and its environment?

MR. RINKER: Mike Rinker, for the record. There will be ongoing monitoring, but I would like to emphasize that we do not see the need for ongoing monitoring to ensure protection of the environment or people, we are confident that that has already been achieved over the last couple of years.

However, we have a number of other facilities, nuclear power plants, SRBT in Pembroke, where our understanding of how tritium is going to continue to cycle over the environment, that knowledge would benefit us from a regulatory perspective.

So the CNSC, together with the Ministry of Environment, will continue to do some monitoring. This will help us disseminate information, it will help us learn about tritium cycling, although I would emphasize it is not necessary to ensure safety we are convinced.

MEMBER VELSHI: And my last question was more the decommissioning cost and how did those compare with whatever financial guarantee there was for decommissioning?

MR. RINKER: Mike Rinker, for the record. So there was a financial guarantee for this facility, exact figures I'm not certain, it was about $650,000 and they used up the majority of that money for their cleanup activities.

So Shield Source did pay for this through their financial guarantee.

MEMBER VELSHI: But there was sufficient funding for it?

MR. RINKER: Mike Brinker, that's correct.

MEMBER VELSHI: Thank you.

THE PRESIDENT: Any other questions? Just one question. When you release it now for industrial use of any purpose, I want to understand, so who -- the local government and the local land-use people and the Ministry of Environment, are they all okay that everything was cleaned up?

MR. RINKER: Mike Rinker, for the record. So there are two stakeholders, the Municipality of Peterborough who actually owns the airport, they are the landlord.

We have been in constant contact with them. They are interested in having information because from a municipality perspective they want to be prepared should any of their constituents have questions and they would like some assurance that the facility is safe, so something documented, so we have agreed to issue them a letter and the Record of Decision would also help to inform them.

I think really their main interest is they want to make sure that they can inform the next tenant and they would like to have another tenant.

The Ministry of Ontario has some concerns on environmental monitoring and so they want to work with us. We are going to meet them again in April about what -- we sent them our preliminary plan to better understand tritium in the environment, but they don't have any regulatory requirements, but they are with us on the table to talk about monitoring in future.

THE PRESIDENT: Okay. Anything else on this? Okay, thank you. Thank you very much.

5.1 Update on the incident involving four uranium hexafluoride cylinders at the Port of Halifax

THE PRESIDENT: The next item on the agenda is an update on the incident involving for uranium hexafluoride cylinders at the Port of Halifax.

I understand that we have representatives from RSB Logistics joining us via videoconferencing for this item.

I understand Mr. Eckel will be connected with us. Mr. Eckel, can you hear us?

MR. ECKEL: Yes. It's George Eckel, for the record. I can hear you well.

CMD 14-M19

Oral presentation by CNSC staff

THE PRESIDENT: Okay, thank you.

I understand that Monsieur André Régimbald will make the presentation from CNSC staff as outlined in CMD 14-M19.

Monsieur Régimbald, vous avez la parole.

M. RÉGIMBALD : Merci, Monsieur le Président. Rebonjour, Mesdames et Messieurs, Membres de la Commission.

I will do the introduction in French and then I will continue the presentation in English.

Alors, aux fins du registre, je m'appelle André Régimbald. Je suis le directeur-général responsable de la réglementation des substances nucléaires.

Je vous présente mes collègues :

- M. Sylvain Faille, directeur de la Division des autorisations de transport et du soutien stratégique; et

- M. Martin Thériault, agent de transport dans la Division de monsieur Faille. J'aimerais ajouter que monsieur Thériault est aussi un inspecteur.

D'autres membres du personnel de la CCSN sont présents dans la salle et peuvent répondre aux questions au besoin.

On March 13, 2014 at approximately

10:15 p.m. the CNSC received a call from Transport Canada's Emergency Call Centre, or CANUTEC, that there had been an incident at the Port of Halifax involving a shipment of uranium hexafluoride or UF_6.

The incident occurred at around nine o'clock that evening when a flat rack carrying four cylinders filled with UF_6 were being offloaded from a sea vessel that was accidentally dropped -- I'm sorry, not the sea vessel, but the rack was accidentally dropped back aboard the vessel from a height of approximately seven metres.

The shipment consisted of four cylinders containing UF_6, which is a fissile material since it is composed of low enriched uranium at a quantity not exceeding five percent uranium-235 enrichment. And please note that this is not weapons grade material as was initially reported in the media.

The UF_6 inside the cylinders is in solid form and maintained under negative pressure. Each cylinder is approximately two metres long by 80 centimetres in diameter and can hold up to 2,277 kilograms of UF_6.

The material originated from Chester, England and was destined to the Westinghouse fuel fabrication facility in South Carolina, United States, to produce fuel for use in light-water commercial nuclear reactors.

Under the CNSC *Packaging and Transport of Nuclear Substances Regulations*, which align with international transport regulations established by the International Atomic Energy Agency, this type of shipment requires the use of certified packages, as well as a CNSC licence to transport while in transit.

The transport while in transit is defined as a shipment being transported through Canada in a situation where the place of loading and the final destination are outside Canada.

This licence, as with any other transport licence, is issued by the CNSC Designated Officer. The package containing the UF_6 cylinders is also certified by the CNSC Designated Officer as meeting the requirements specified in the PTNS Regulations and the international IAEA Transport Regulations, after a thorough technical assessment performed by professional engineers at the CNSC.

When preparing the shipment for transport, each cylinder is inserted inside an overpack, in this case it is a model UX-30 Overpack, which consists of a stainless-steel casing filled with polyurethane foam which serves as a thermal blanket for the UF_6 and protects the cylinder from shocks during transport. The package is therefore comprised of the cylinder and the overpack, and was certified by the CNSC Designated Officer on June 23, 2011 as an endorsement of the US Department of Transport Certificate, issued under a similar process, and used worldwide.

On the next slide -- and please, I apologize, I should have indicated before the presentation that you have two copies of Slide 6 in your deck, so Slide 7 is going to be a new slide, and I'll explain later.

Also, I should have mentioned that -- I'm sorry, Slide 20 has a better photograph that will illustrate the object of the slide.

So resuming -- we're on Slide 6 now -- the company responsible for the shipment involved in the incident is RSB Logistic Inc. It is a transport company located in Saskatoon, Saskatchewan that offers a wide range of transport services including freight forwarding services, and was licensed by the CNSC for this shipment. The CNSC Designated Officer issued the transport licence to RSB Logistic on February 19, 2014, with an expiry date of May 31, 2014.

During transport, the packages were tightly secured to a metal flatrack which consists of a platform with two ends that is used for ease of transport. A total of four UF_6 packages were attached to the flatrack for a total weight of 18,000 kg. Total weight consisted of the four cylinders, their overpacks and the flatrack.

This slide shows the vessel, the *Atlantic Companion*. At the time of the incident, the vessel was docked at the Port of Halifax and was being off-loaded.

This slide shows the crane being used to off-load the packages, which is a 1982 Paceco ship-to-shore crane with a 40 metric ton capacity, or 40,000 kg.

The next series of slides will show how the incident happened.

So, this first slide shows a schematic view of the crane-lifting bridge getting into position to lift the flatrack with the UF_6 packages. The lifting is done by anchoring four attachment points underneath the lifting bridge to each of the corresponding four attachment points on the top of the flatrack. I will show some pictures later on to show you exactly how the anchoring is done.

Before the lifting starts, the crane operator has to secure the attachment points into the anchored position so that the flatrack can be lifted properly.

Then the lifting started, but what happened is that two of the lifting points on one side of the flatrack had not been properly anchored, so all of the 18,000 kg weight was being supported by only one side of the flatrack.

Due to this tremendous weight supported by only one side of the flatrack, that side eventually sheared off from the flatrack as the flatrack swung downward in a pendulum motion to the vertical position, and fell down into the cargo hold from a height of approximately 7 meters.

Immediately after the incident, the Halifax HazMat team was called on scene and the vessel *Atlantic Companion* was evacuated. The HazMat team established a safe perimeter zone of 50 metres around the vessel. This precautionary measure was taken after the HazMat team measured 2 microseiverts per hour at 6 metres away from the packages. This level of radiation was normal for this type of package and consistent with the information contained on the shipping document. The level of radiation measured presented no danger for people working around the packages.

Just to get a sense of perspective, standing 6 meters away from the packages for a period of 10 consecutive hours would result in a dose comparable to one received by a passenger flying across Canada.

Shortly after, RSB Logistic, the licensee, was contacted and immediately implemented its Emergency Response Assistance Plan, and dispatched its personnel to the site. Early on, after being notified by CANUTEC, CNSC transport experts provided initial assistance from our CNSC Headquarters in Ottawa to the first responders and other people who were conducting the incident response on site.

The next slides will provide an overview of how the recovery operation was conducted. So this slide shows the flatrack with the four packages that dropped approximately 7 metres, or 23 feet, back into the ship's cargo hold. As per CNSC and international regulations, this package design has to withstand a drop of 9 metres, or 30 feet, in order to be certified, and the package was certified.

Slide 15 shows, again, the flatrack, as they dropped, so you can see the UF_6 cylinders, which landed on top of another container that was beneath the container of the flatrack. And these show plastic tarps that are above that container.

And you can see that the cylinders are still attached to the flatrack, and the end -- one end of the flatrack is still attached to the base.

These pictures show the sheared side of the flatrack. So you can see, on the lefthand side, the side of the flatrack still attached to the base, and you can see where it sheared off, and the other part was still attached to the crane bridge.

This is a closer look at the sheared-off side of the flatrack which is still attached to the crane bridge.

This is a picture of the sheared-off side of the flatrack after its removal from the crane bridge. And you can see, on the left, the section that was sheared off.

This is a picture of the anchor points on the flatrack, which are essentially female connectors -- see in the red circles. You can see in the enlarged picture on the righthand side that the connectors underwent extreme tensile deformation as these were supporting all of the 18,000 kg of weight being lifted.

These are pictures of the anchor points underneath the crane bridge, as shown in the red circles. The anchor points are male connectors which slide into the female connectors on top of the flatrack at each corner to secure the flatrack under the crane bridge in order to lift it.

On March 14, the morning following the incident, the CNSC sent an inspector, Mr. Martin Thériault, to the Port of Halifax to provide CNSC regulatory oversight of the recovery operation and any potential clean-up activities. Upon arrival, the CNSC inspector boarded the vessel and took additional measurements.

Following his assessment, the inspector confirmed that all of the four packages involved in the incident were intact and that there had been no release of UF_6 from the packages.

On Saturday morning, March 15, the emergency response team, under the guidance of the licensee, proceeded with the removal of the packages from the vessel and storage onto the docks. Once that was completed, the port terminal resumed normal operation.

These are various pictures showing the packages being removed from the vessel and getting ready for storage onto the docks.

From this incident, the CNSC observed that the regulatory system worked as expected, that is:

- The packages contained UF_6 in a quantity not exceeding the quantity specified in the package design as certified by CNSC, and as per the CNSC licence;

- RSB Logistics was duly licensed by the CNSC; and all the required transport placards and labels were in place on the containers and the flatrack;

- The packages withstood the drop without breach as per the package design certificate;

- The first responders took immediate action and contacted CANUTEC, who, in turn, promptly contacted the CNSC;

- The licensee also promptly contacted the CNSC, as required by regulations;

- The licensee responded effectively in implementing its emergency response plan as per the licence and complied with regulatory requirements.

In conclusion, the incident presented no risk to the workers, the public and the environment.

Radiation emitted from the packages remained at the expected level for normal transport for this type of package, it was perfectly safe to work around the packages, there were no breach of the packages, there were no spills of UF_6, there were no security concerns, and there were no injuries.

On Monday, March 24, four new UX-30 overpacks were sent to the Port of Halifax to re-package the UF_6 cylinders and move them to destination.

On Tuesday, March 25, the packages involved in the incident were opened and a visual examination of each cylinder was performed by specialists who had been dispatched on site by the consignor and the consignee, including a representative of the American National Standards Institute, ANSI, as the cylinders are designed and manufactured according to ANSI N14.1.

The cylinders were declared free of damage and were repackaged inside the overpacks.

The packages were reloaded onto a new flatrack. At that point, the CNSC inspector was satisfied that all regulatory requirements were met in order for the packages to resume their transportation to the United States.

In closing, RSB Logistic will be submitting a report on the incident to the CNSC by April 3, 2014. What we understand so far is that the incident is the result of one side of the flatrack not being properly secured to the crane bridge prior to lifting the flatrack.

As mentioned before, this resulted in all the 18,000 kg weight being supported by only one side of the flatrack. This ended up in the unsecured side moving downward in a pendulum motion towards a vertical position and caused the attached side to shear off from the flatrack base.

The cause and circumstances of the incident are still under review. We do not know at this time if the incident was the result of a mechanical failure -- that is, a problem with the anchoring mechanism -- or human error -- that is, the operator not ensuring that the connectors were secured into their anchoring position.

CNSC staff proposes to follow up with the Commission in writing in due course, with any additional information that may become available regarding the cause and circumstances of the incident.

Thank you very much.

THE PRESIDENT: Thank you very much. I'd like to open the floor for questions. I'd like to start with Monsieur Tolgyesi.

MEMBER TOLGYESI: Merci, Monsieur

President.

When you're looking in the port, usually the speed of loading/unloading containers is of the essence. They should move lots of stock, containers, and other material. And usually it's mechanical, I suppose, and it's anchored and after it's unhooked automatically or by ... And is this type of accident frequent, to your knowledge, in the ports? Because they handle thousands and thousands and thousands of containers. So I expect it was not necessarily the first or I hope it's the last.

MR. RÉGIMBALD: Andre Régimbald here. To our knowledge this is the first incident that has been reported to the CNSC, but perhaps if I can ask the person from RSB Logistic to provide information about the number of such operations in the port.

THE PRESIDENT: Mr. Eckel, over to you.

MR. ECKEL: George Eckel, RSB. We are told that it happens approximately once every 200,000 container moves is the information we've been told at this time.

THE PRESIDENT: From whom? Who

told you that -- where is this information coming from?

MR. ECKEL: This is information directly from the Port of Halifax.

MEMBER TOLGYESI: So when you say I don't know how many containers the Halifax port is handling, but if it's every 200,000, that means there is -- I suppose they handle more than 200,000, so it's several times a year it happens. Do you have independently and what was in this incident/accident involved, it was hexafluoride, but for other material do you have any maintenance, inspections tests and the frequency of procedure revisions?

MR. ECKEL: I'm sorry, could you clarify the question? I guess we have regular maintenance inspections, recertifications for our owned equipment, the flatrack. But as far as the port equipment, no, we do not.

MEMBER TOLGYESI: Okay. Then it was mainly for port facility because that's their equipment, it's not yours?

MR. ECKEL: Yes, that's correct.

MEMBER TOLGYESI: And you -- to your knowledge, you don't know what's the procedures and what's the frequency of maintenance of this equipment?

MR. ECKEL: We have asked for the maintenance records and service records of equipment and are still waiting for that.

MEMBER TOLGYESI: Okay. When you look, it's -- does the cargo hold contain only one or several layers of flatracks?

MR. ECKEL: Several layers.

MEMBER TOLGYESI: Could you tell me what's the maximum lift of container, more specifically flatracks, when loading or unloading? Because here it was a 7 metre drop into the vessel cargo hold on the other layer. Is it possible that at one point the potential drop will exceed 9 metres as you move it up and you move it to the dock, and that potential drop will exceed 9 metres, therefore, overpass the regulation design drop limit?

MR. ECKEL: If I could pass that question to one of my colleagues, Mr. Klassen. As far as the positioning of the containers on the vessel, he could comment on that.

MR. KLASSEN: At this time I would not be able to give an exact answer on that because that's something I am not familiar with. I'm not on the terminal. I don't know the operations of a terminal and the height that they lift. I have the same question and -- but at this time I don't have an answer.

THE PRESIDENT: Can I piggyback on this. What I'd like to know is whether there are specific instructions or regulations on -- on this particular material that indicates to the Port Authority thou shall not lift this box above 9 metres? Is there such a regulatory or procedural requirement?

MR. RÉGIMBALD: I will ask Mr. Sylvain Faille to provide information on that.

MR. FAILLE: Sylvain Faille for the record. The regulations don't require any -- or there is no limitation on the height for the lifting. The regulations were designed for transport purposes. And the 9 metre drop, which is specified in the regulations for those packages, is to simulate a road accident at a certain speed. But the other important part is the target where the package has to be dropped onto, which is defined in the regulations as an unyielding surface, which is extremely hard compared to a road or even concrete. So there is some compensation there in terms of the drop height as 9 metres, as a set level, but the surface onto what it has to be dropped onto is much more stronger in the regulation than what is happening in normal accidents. But to answer the question, there is no limit on the height when it's lifted. It's not 9 metres or anything like that.

THE PRESIDENT: Monsieur Tolgyesi

MEMBER TOLGYESI: Because potentially what you are looking here, if it was not a last row or bottom row of these flatracks, the drop could be higher than 9 metres. And when you lift it above the side of a ship and you move it, I think that it could be even higher. So I think we should look that, what will happen if it's -- we should review this.

The other question is, I have, could you compare the container certificate? When you compare -- you know, we had a large accident in Eastern townships, where the train just blew up. And the question there was questioning the container's quality and resistance. So could you compare what's the certification of this hexafluoride containers versus railroad fuel and other liquid containers?

MR. FAILLE: Sylvain Faille for the record. I don't have the exact information on to railcars or tanker for other type of dangerous goods, but for the radioactive material all the requirements are in the regulations and, like I said, they have to resist accident conditions, which is simulated by the 9 metre drop test that we just mentioned, which is one of the tests. And on top of that, they also have to withstand a thermal test for about 8-- it's 800 degree celsius for half an hour. And there is also immersion test following that. So there is design requirements in the regulation that each package that requires certification needs to meet before they are approved. And I am not sure of the design requirements for railcars or other -- package design for other dangerous goods.

MR. RÉGIMBALD: I should note also - it's Andre Régimbald here - that the UF_6 itself is very different from the commodity that was transported involving -- involved in the Lac-Mégantic accident. And I'll ask Mr. Michael Rinker to provide just an overview of the physical and chemical characteristics of UF_6 as it is packaged.

MR. RINKER: Mike Rinker for the record. The Port Hope conversion facility makes a very similar product and stores it and transport it in similar ways. The product UF_6 once it cools to normal ambient temperature is a solid. It's not flammable on its own. It can -- it's an oxidant, so it can make other things burn. But I guess the -- one of the major risks for exposure to UF_6 would be more of a chemical nature. Even the uranium itself is more of -- its risk is more of a heavy metal than it is from the dose perspective. And if it gets wet, then fluoride can combine with water and other things and be quite corrosive to the skin. So its risks are associated with more of a strong chemical hazard, but it is -- in the form that it's being transported in, it's in a solid form. So if the container were to break, it could be cleaned up. If it were to get wet, then it's a little bit more difficult.

MEMBER TOLGYESI: These cylinders are not under pressure or under negative pressure?

MR. RINKER: Mike Rinker for the

record. They're under negative pressure.

MEMBER TOLGYESI: So if an accident happens for any reasons, it could -- the containment could blow out? When I say "blow out" I --

MR. RINKER: Actually, I think it's the opposite. If it was under a positive pressure and it breached, it would blow up. Under negative pressure it would be the opposite.

MEMBER TOLGYESI: And these cylinders, what's the life of these cylinders and what's the life of the overpacks? And how -- the overpacks are certified also? Because we have a certification for the cylinders. But how is the overpacks are certified? And when you combine them, because that's when -- you know, when there is drop we could see what will happen.

MR. FAILLE: Sylvain Faille. Actually, what the -- is certified is the overpack with the cylinders, that's the package, and that is what is certified under the CNSC. The cylinders themselves also need to be approved under the ANSI standard, which is a different one and that only combines for the -- looking at the cylinders themselves. But on our end, we're looking at both, the overpack plus the cylinder, that's the package, and that is the combination that we are certifying. And as part of that, they have regular maintenance on -- done on those packages, including the cylinders and the overpack. And that is under regular maintenance. Perhaps RSB Logistic could provide more information on actually the frequency of inspections.

MR. RÉGIMBALD: And the certification process is a rigorous technical review of the design of the package and cylinder, and the review is performed by accredited professional engineers at the CNSC, who follow the ANSI -- review against their ANSI standards and other industry standards and based on expert judgment.

THE PRESIDENT: Okay. There's going to be another round, so let's move on to Dr. McEwan.

MEMBER McEWAN: Thank you, Mr. President. Can I just be clear about the sequence. The accident happened, the first responders would notify the company RSB and the CNSC or did I hear that you said RSB was notified and then CNSC was notified?

MR. FAILLE: Sylvain Faille. I'm not sure exactly the order, the exact order, but I know the first responders called CANUTEC, which is the service -- emergency service from Transport Canada. And RSB Logistic was also notified. I'm not sure who was notified first on the two, but it was probably very close one to the other. From the CNSC, we were notified through the Transport Canada CANUTEC service, who have a direct line with our duty officer.

MEMBER McEWAN: So there wouldn't be an expectation for a radioactive package that CNSC would be notified as one of the first notifees -- people to be notified?

MR. RÉGIMBALD: It could happen, but normally the first responders are trained, when there is a transportation accident, to contact CANUTEC first and then CANUTEC would contact us, and this occurs very rapidly.

MEMBER McEWAN: And the time between CNSC being notified and the inspector being on site was how long?

MR. RÉGIMBALD: I'm just looking at my notes. We were notified on the evening of

Mar-- well, Sylvain, Martin, can you -- Martin Thériault will provide the exact detail.

MR. THÉRIAULT: For the record, Martin Thériault, transport officer with CNSC. We were notified by CANUTEC through the duty officer on the Thursday evening at 10 o'clock and was on site at 4:00 p.m. the following day. I was in Halifax.

MEMBER McEWAN: That seems a long time.

MR. RÉGIMBALD: The situation was under control through the night. The first responders established a safe perimeter around the vessel. At that point it was difficult to see if there was any leakage from the containers, but the radiation level measured at 6 metres from the packages indicated that the radiation level was normal. Once we got in on the Friday morning, we organized ourselves to dispatch Mr. Thériault to Halifax and the first flight that he could hop onto was at 1:00 p.m., I believe, or there, around that time from Ottawa. So as soon as he got on site, then he immediately went to the port.

MEMBER McEWAN: I still think, Mr. President, that seems a slow response time to get somebody from CNSC on site. I mean, it must have been possible for somebody to get there much sooner than that flying through Toronto. I mean, there must be flights in the morning.

MR. RÉGIMBALD: The situation assessed, again, was -- was under control at the port. There was no indication of someone's life was at stake. We did our due diligence and obtained exactly the information and started to organize ourselves early on Friday morning. So by the time we organized the travel, as I said from Ottawa, the first flight was at 1:00 p.m. to Halifax.

THE PRESIDENT: But you know that in the first few hours there was a bit of misinformation, that we were not sure whether there was a leak or not, and the dose reported -- or put the scare of people because they were talking about four times background -- whatever that meant. So I think the question is what would happen if there -- how do you know it wasn't at that time something to worry about?

MR. RÉGIMBALD: Perhaps I can clarify that. We were -- we were assisting from -- remotely from Ottawa early on. So we were providing assistance to the people on site who were conducting the first response. We would get worried if radiation levels were above what it would be normal for this type of transport. Or if there was a spill, a confirmed spill. And as I said, we are involved early on to assist first responders until we organize ourselves to dispatch physically an inspector on the scene.

THE PRESIDENT: Monsieur Awad.

MR. AWAD: Raoul Awad, Director General of Security and Safeguards. Since -- we had been informed at 10:00. At 10:15 there was a direct line with the first responders, what -- the measurements they took, and at that time we start assessing what kind of measurements and, if needed, immediate support to the first responders. And we are contacting Mr. Thériault, and we're contacting at the same time our inspectors in New Brunswick, in Saint John. And the choice was either we will send somebody from Saint John or from Ottawa, but due to the radiation level, which is very low and the situation is totally under control, the decision was to ask Mr. Thériault to go the next day instead of sending during the night somebody from Saint John.

THE PRESIDENT: Thank you. Dr. McEwan. Okay.

Ms Velshi.

MEMBER VELSHI: I've heard in the media that there have been similar incidents in the past, maybe one such incident in Halifax and perhaps something a bit more serious in the U.S. Can you provide some more details on that, please?

MR. RÉGIMBALD: André Régimbald here. We're not aware of other similar incidents at the Port of Halifax. I'll have to examine or review the information that was published in the media, but to our knowledge this is the first incident of this type that happened in any port in Canada.

MEMBER VELSHI: Okay. So I heard there was something in the 1990s and then in the 1980s in the United States that was a little more serious, but maybe I'll pass it on to Marc where I had heard that and maybe you can follow up on that.

THE PRESIDENT: But is RSB aware of any incidents specifically on this type of shipment in the U.S. or anywhere else?

MR. ECKEL: George Eckel for the

record. No, I am not aware of any similar incidents with similar cargoes.

THE PRESIDENT: Okay. Thank you. Ms Velshi.

MEMBER VELSHI: On slide 24, where you talk about how all the things worked, have there been areas of improvement that you have identified, things that could have worked better? We have heard about some early miscommunication and it seems like there is still a bit of public angst about was there a leak, why were the dose rates higher. So if you can maybe shed some light as to, you know, what are some of the learnings from here and how could things have been better communicated in particular?

MR. RÉGIMBALD: We are -- we have scheduled a lessons learned meeting on April 4th with all those who were involved from the CNSC side, and we will be able to determine at that time what worked and what didn't work, and what we can improve. So we can provide further information at a later time on that.

MEMBER VELSHI: Thank you.

THE CHAIRPERSON: And so will RSB provide a report on this. Is that correct?

MR. RÉGIMBALD: Yes, RSB indicated to us that they would be submitting a report on April the 3rd.

MEMBER VELSHI: You used new Overpacks. Was that just a precautionary measure, or was there some evidence that the existing ones had been damaged?

MR. FAILLE: Sylvain Faille, for the record.

I think the main reason was to also there was some damage to the Overpack, but there was not extensive damage. But they -- the fact that some of the saddles were torn and things like that was much easier to bring in a new flat rack with new Overpacks and make sure that everything was done and then they could verify each cylinder before moving on to their final destination was a precautionary measure.

MEMBER VELSHI: Thank you. I'll come back in my second round.

THE PRESIDENT: Monsieur Harvey?

MEMBER HARVEY: Just one question about the -- you mentioned that the Overpack has to meet standard and -- but what about the cylinder?

I mean, even if the -- if the Overpack -- if we suppose the Overpack broke at that time, this is not to say that would be radiation. So the cylinder itself has a certain resistance, I suppose.

MR. FAILLE: It's Sylvain Faille.

That is correct. There is some testing required -- requirements for those cylinders in the ANSI Standard, including some pressure. They also have to meet the requirements for when they are being filled in a plant or when they are being emptied, so those ones -- and they are also -- and the standard applies for when the cylinders are being manufactured and insured that they would be safe to handle in a facility where they have to be lifted also and as well as filled and emptied.

But I can't -- I'll have ---

MEMBER HARVEY: What could happen with the cylinder if you drop it when trying to put it in the Overpack, to insert it in the Overpack?

If you drop the cylinder, what could happen?

MR. RÉGIMBALD: This is -- the

loading in the Overpack occurs within the facility. Mr. Rinker might have some additional information or, otherwise, we would get back to you on that.

MR. RINKER: Mike Rinker, for the record.

I have to get back to you on that one.

MEMBER HARVEY: Okay. Thank you.

THE PRESIDENT: Thank you.

Any other questions? Anybody else? Ms. Velshi.

MEMBER VELSHI: You mentioned the first responders get trained. Who trains them?

MR. AWAD: Could you repeat the question, please?

MEMBER VELSHI: Who trains the first responders?

MR. AWAD: We train the first responders for emergency nuclear and radiological emergency. And we give training for the first responder all across Canada under the CBRNE program, which is the Chemical, Biological, Radiological and Nuclear program.

MEMBER VELSHI: And they would be

the ones responsible for measuring those rates and communicating their findings?

MR. AWAD: That's correct.

MEMBER VELSHI: So that will be part of your review as well, then, on how well that happened.

The ---

THE PRESIDENT: So just to conclude on this question, so are you happy with -- I think the first responder was the fire authority in the port, if I understand correctly. And those are the people we're talking about?

MR. AWAD: That's correct, yeah. The HAZMAT team at ---

THE PRESIDENT: Right. And the fact they did a good job on everything besides communicating with the press.

So are you satisfied that they actually ---

MR. AWAD: Actually, they did exactly what we expect them to do. They are the first responder. They are the first to be on the scene of that accident, taking measurements and inform the authority.

THE PRESIDENT: I know, but when

I -- I thought one of the first observation is to assess whether there is safety and risk to the public. And I don't think they've -- in fact, everything they've said about the dose and the hazard was not, let's put it this way, assuring the public when they didn't have all the information.

MR. AWAD: Actually, they -- when they give to the media, they give it as relative to the background radiation, which is not the right way to do it. And that's something we need to maybe adjust in our training.

THE PRESIDENT: Ms Velshi.

MEMBER VELSHI: My last question was around port operations.

In slide 25, you say it was perfectly safe to walk around the packages, and I -- and this determination was made when the inspector went on site the next day.

So why did it take the port a couple more days before it resumed operations?

MR. M. THÉRIAULT: Martin Thériault, for the record.

The decision was made by the terminal authority or terminal operator based on the assessment done by the radiation expert that was hired by them, so they wanted him to confirm that no level of radiation were detected on the surface of the container. And that's when they had sufficient insurance that there were no loose contamination on the surface on the container.

And also, they wanted to safely remove the container from the cargo hold before to resume operation because it was still a small risk of an incident happening because the cylinder were not affixed the way they were normally be affixed to the container -- to the flat rack. So they want to mitigate the risk having less people go around when there's still an emergency -- an incident scene on board the ship.

MEMBER VELSHI: Thank you.

THE PRESIDENT: Okay. Dr. McEwan?

MEMBER McEWAN: Thank you, Mr. President.

This is a question of naiveté, so please bear with me.

On slide 7 and slide 21, I think I'll refer I'll to as well, the -- if I understand it correctly, the package is put on the flat rack and the flat rack is what is installed in the hold?

MR. RÉGIMBALD: Yes, that's correct.

MEMBER McEWAN: Is the package bolted to the flat rack? Because in slide 21, it looks as if, with the fall, they've all rolled off and down towards one end.

MR. RÉGIMBALD: Yes, it is bolted to the flat rack. We have pictures of that.

Apologies, we didn't include them in here, but perhaps Mr. Faille or RSB Logistics can add further detail.

MEMBER McEWAN: So did the bolts -- did the bolts hold?

MR. RÉGIMBALD: Yes. It's -- it has to be securely bolted to the flat rack.

As you can see on the top picture on slide 7, you see the supporting things there, so the base of those supporting racks are bolted onto the flat rack base.

MEMBER McEWAN: So in slide 21 where you see the package, that is still bolted to the base of the flat rack.

MR. RÉGIMBALD: Yes, correctly.

MEMBER McEWAN: Okay. So just

intuitively, would it not be safer and more consistent for the flat rack to be in a container?

I mean, I just happened to be watching some containers being removed from ships the other day. I was on vacation in good sight of this.

They're taken a long way in the air. There's a very, very rapid turnaround, and clearly speed is the whole purpose of this.

Doesn't it add a risk just to have the flat rack bolted onto the crane? Wouldn't it be safer and more reassuring if the flat rack was in the container because then you've got continuity of process for the port operator?

I -- that's a naïve question born of process ignorance.

THE PRESIDENT: Maybe RSB can shed a light on this.

MR. ECKEL: George Eckel, for the record.

Just to clarify, a flat rack basically is a shipping container, same as a sealed container, just simply has no roof or no sides. And the reason they select a flat rack is simply for access for loading the cylinders and things like that.

But it has the identical anchor points top and bottom, same dimensions as a closed sea container, if that answers your question.

MEMBER McEWAN: Thank you.

And presumably, any time a flat rack is loaded, it would always be in the hold. It wouldn't be above the level of the deck.

MR. ECKEL: That is correct.

MEMBER McEWAN: Thank you.

THE PRESIDENT: Monsieur Harvey?

MEMBER HARVEY: I had a question. I don't know if you just answered -- just asked that question at the end. But my question was, is there any indication where to place the flat rack in the boat? It has to go in the hold or it could be placed on the deck, anywhere on the boat?

MR. KLASSEN: (Off mic) In the hold.

THE PRESIDENT: I think we just got the answer, in the hold. That's what they said.

MEMBER HARVEY: Is it an obligation to ---

MR. RÉGIMBALD: There is no

requirement in our regulations to that effect but perhaps the licensee can provide information.

THE PRESIDENT: RSB?

MR. ECKEL: Yes, there -- George Eckel, for the record.

They're always in the hold. That's a commitment from the ship lines to do that.

MEMBER HARVEY: And you can have two or three flat racks, one over the other?

MR. ECKEL: Typically, there is separation between these flat racks or containers of other cargoes in between each one. They're not typically put together.

MEMBER HARVEY: Thank you.

THE PRESIDENT: Okay. Anybody else?

You did say you're going to do kind of a lesson learned report on all of this, so we look forward to reading that.

The good news was that there was no leak and there was no breach, but what the concern is, you know, we always like to ask the "what if" question. If there were, what would we expect different behaviour here, so we look forward to reading this.

Thank you.

I'd like to take a 10-minute break now, and so we'll reconvene at 11 o'clock.

--- Upon recessing at 10:43 a.m. /
Suspension à 10 h 43

--- Upon resuming at 11:06 /
Reprise à 11 h 06

5.2 Presentations on Fitness for Service of Pressure Tubes

THE PRESIDENT: Okay. Sorry we're a bit late here.

And now for something completely different, I understand. The next item on the agenda is on fitness for service of pressure tubes.

And I understand that there's going to be a presentation by OPG and Bruce Power. It's a joint presentation as outlined in CMD 14-M15.1. And I understand Mr. Saunders will make the presentation.

Please proceed.

CMD 14-M15.1

Oral presentation by Bruce Power and Ontario Power Generation Inc.

MR. SAUNDERS: I will start, at least.

So good morning. Frank Saunders, for the record, and thanks for the opportunity to present this. I think it's a good information session.

And we're here to discuss pressure tube fitness for service. It's come up in the past, and we want to provide you some information related to that.

We know that pressure tube material properties change with time and with service, as do most materials, actually. And we've engaged as an industry -- this is a joint effort that we do -- in looking at the material properties of the pressure tubes and how that will change with time, especially going into the future, and looking at what those changes -- what impact those changes might have on operation.

So we're here this morning to talk

about how we manage that fitness for service and on the R&D programs that we have in place to give us that predictability into the future.

I have two people with me who are much more knowledgeable on this subject than I, and they'll do most of the talking. So on my immediate left is Dr. Paul Spekkens, who is Vice-President, Science and Technology Development, for OPG. And on my far left Mr. Gary Newman, who's a Senior Vice-President and Chief Engineer at Bruce Power.

And Gary will start with the presentation.

MR. NEWMAN: Gary Newman, for the record, and good morning.

First and foremost, and by way of introduction, we are confident that our reactors are safe to operate into the foreseeable future. This confidence stems from a combination of the next two bullets.

The pressure tube life cycle program is a mature, well-defined and managed protocol which is aligned with all licensing requirements. And secondly, the program incorporates a continuous improvement philosophy --

--- Pause

MR. NEWMAN: Okay. So maybe I'll return to the beginning.

Okay. By way of introduction, and first and foremost, we are confident that our reactors are safe to operate into the foreseeable future. This confidence stems from a combination of the next two bullets on the introductory page.

The pressure tube life cycle program is a mature, well-defined and managed protocol which is licensed with all licensing requirements. And secondly, the program incorporates a continuous improvement philosophy, ensuring a state-of-the-art understanding is maintained.

Let me now walk you through the key components of our presentation today, beginning with the building blocks for continued safe operation, followed by a brief description of the CANDU Reactor and, specifically, the pressure tube design.

This will be followed by a high level summary of the pressure tube life cycle management process, and then I will turn it over to Dr. Spekkens to describe some of the more recent research and development results.

Output from these results have been used to update the pressure tube fracture toughness modelling.

This fracture toughness modelling, in turn, is used to divine our operating envelope and is also used as input to leak-before-break assessments.

By the rules of the Canadian standard, we are allowed to use both deterministic and probabilistic leak-before-break assessment methodologies. The probabilistic leak-before-break approach will be the focus today, and this will be discussed by Dr. Spekkens as well.

And of course, one needs to compare and continue to confirm low risk of pressure tube failure.

And finally, Dr. Spekkens will cite some conclusions and provide an overall summary to the presentation.

Moving, then, on to the building blocks for safe operation, I would begin by indicating that the original designers utilized robust design assumptions which led them to the target life of 210,000 Equivalent Full Power Hours. This is approximately 30 years in duration, assuming an 80 percent full power operation, whereas since the units have been in service, the practice is to utilize fitness-for-service assessment methods based upon a combination of more than 30 years of research and development and an accumulated 500 reactor years of operating experience.

This is supported by a solid life cycle management protocol for pressure tubes, which is patterned after the IAEA guidelines and their application.

And if one looks in aggregate at the fitness-for-service measurements over the operating life, it shows pressure tube material properties exceeding original expectations.

Detailed guidance for life cycle management requirements are captured with -- mainly within two Canadian standards, CSA N285.4 and CSA N285.8.

Detail guidance for life cycle management requirements are captured within these standards, and the output from these assessments are used to confirm fitness for service into the foreseeable future.

In keeping with our continuous improvement philosophy, work will continue to enhance the existing understanding and further underpin the excellent performance observed to date.

Now moving to the next slide and to a high level description of the reactor design, specifically as it pertains to the pressure tubes, the photo in the overhead shows a normal reactor face with a collection of pressure tubes, and also shows the network of feeders.

CANDU reactors have multiple horizontal pressure tubes and, in the case of Bruce and Darlington designs, 480 in total, whereas for Pickering and the CANDU 6 designs, there's 380 pressure tubes.

The pressure tubes contain both the fuel bundles and heavy water necessary to transfer heat to the steam generators.

I should note that the zirconium pressure tube alloy is very similar to the more widely-known family of titanium alloys. Both of these alloy systems are known to be high quality, corrosion-resistant materials which have served us well throughout their life.

Perhaps now, in the next slide, a bit more detail on the pressure tube design which, as shown in the cut-away schematic, is comprised of the fuel bundle surrounded by the pressure tube which is supported by the yellow spacers, annular spacers, in the diagram.

This, in turn, is surrounded by a calandria tube. And the assembly which contains the pressure tube is surrounded overall by the moderator within the calandria vessel.

Dry gas flows around the outside of the pressure tube and moves and provides moisture detection capability.

This moisture detection capability supports a portion of the fitness-for-service methodology, specifically, the leak-before-break methodology that Dr. Spekkens will speak to in his set of slides a bit later in the presentation.

Now moving to the next slide, to the -- and on to the pressure tube life cycle management process depicted in the diagram, we follow the "plan, do, check, act" approach to process implementation beginning, of course, with an initial plan at the top, 12 o'clock position.

This plan provides guidance on how to best operate, monitor and remediate our fleet of pressure tubes, and it does so in a manner which minimizes the expected effects of component aging.

So examples of this operating framework would be primary heat transport system cleanliness, chemistry control and the envelope within which we ensure that our pressure tubes operate.

Next, we monitor the condition of our pressure tubes through planned outage inspections and surveillance activities, an example of which would be scrape specimens, which are removed and analyzed to monitor for hydrogen isotope concentrations.

Based upon this monitoring, we derive our understanding and account for any observed changes which may prompt additional maintenance. So for example, we may need to locate and relocate the support spacers that we talked about earlier in the schematic and other forms of remediation such as further research and development.

And finally, we have some -- we have come full circle and now must update the life cycle plan to take on the new information, both positive observations and areas for improvement.

So now moving on to the next slide, a brief discussion about the manner in which our pressure tube life cycle management process is captured within the Canadian standards.

First of all, ongoing monitoring of the fuel channel condition is captured in CSA N285.4 to ensure pressure tubes remain safe to operate. This is inclusive of periodic and inspection -- in-service inspection as well as material surveillance.

These observations are then evaluated utilizing standard fitness-for-service assessment process and criteria. These are defined in CSA N285.8.

This includes assessment of known and projected conditions that the fuel channel will see or pressure tube will see. It involves both individual pressure tubes as well as the aggregate core evaluations. It also includes evaluation of material properties and any observed changes in those properties.

And Dr. Spekkens will speak in more detail about what we have observed in the case of our pressure tube fracture toughness in the next series of slides.

Moving on to the next slide, the fitness-for-service philosophy further supports the series of reactor design barriers that are in place to ensure safe, reliable operation of our reactors.

As you can see, represented by the first barrier, we complete R&D as well as monitor pressure tube performance, which supports the assessment of the current pressure tube condition.

We combine this information with inspection results from barrier 2, which are subjected fitness-for-service assessment to confirm solid performance over the next operating interval. This demonstrates no unacceptable aging effects over the future operating period.

Barrier 3 is a further defence in-depth which postulates that frets exist which have not been detected as part of the requirement to demonstrate leak before break. This capability provides operators with the ability to be notified and, by following our operating procedures, place the reactor into a safe shutdown state.

The fourth barrier represents a safety system designed into the plant which would activate automatically and similarly shut down the unit in a safe and timely manner should the operator fail to do so.

The fifth barrier is in place as a further source of cooling water from our emergency cooling system, which again protects the fuel to ensure proper cooling.

And as a final barrier, barrier number 6 represents the reactor containment structure, which prevents the release outside of containment.

I'd like to now turn it over to Dr. Spekkens to provide you with an update on the R&D activities over the last three or four years and the assessment methodologies which are -- which utilize these results as input to our fitness-for-service assessments.

Dr. Spekkens.

DR. SPEKKENS: Thank you, Gary.

For the record, my name is Paul Spekkens. I'm Vice-President of Science and Technology Development at Ontario Power

Generation.

And I'm going to use this overview slide as a road map to indicate when we're moving from one topic to the next as we go through the presentation.

So as Gary indicated, I'm going to start by talking about some of the research and development that we've done in response to the changes that we see in the properties of our pressure tubes.

So material properties are important to safe operation of the pressure tubes, and part of the standard fitness-for-service process that we follow is to monitor for changes in material properties as pressure tubes age.

We know that hydrogen concentration is the most important influence on fracture toughness of our pressure tubes, and fracture toughness in the context of what I'm going to be speaking about simply means the ability of a material to resist the propagation of a crack through it.

If a material resists the progression of a flaw, then it's said to have a high fracture toughness.

Now -- so hydrogen is an important influence, and we know that hydrogen concentration increases with time that a pressure tube is in service. And so we've been -- we've undertaken an R&D program to examine the impact of future hydrogen levels on the toughness of our pressure tubes.

So we've been doing R&D to look at the effect of hydrogen on the resistance of pressure tube material to crack propagation as you're putting internal pressure in the tube. And that's -- and we measure this property of fracture toughness by doing burst tests. And that's what I'm going to illustrate on this slide.

So we do a burst test by pressurizing a length of pressure tube removed from one of our reactors, typically about 18 inches long. Now, these pressure tubes that we remove from our reactors are highly radioactive, so all this work has to be done inside a hot cell. And the photographs that you see on this slide are a little bit blurry because they're taken through the window of a hot cell at the Chalk River facility.

We start with a tube section, the

photograph on the left, and then we have to put a very substantial starter flaw in it.

This flaw is a bit more than two inches long, much, much bigger than anything we'd ever see in a reactor. And the reason we have to do that is because if there isn't a starter flaw, you can't burst one of these pressure tubes within the allowable operating pressure range.

So to do one of these tests, we put the starter flaw in, we raise the pressure inside the section of pressure tube until eventually the flaw starts to propagate rapidly.

And that's the photo on the right, which shows that the starter flaw which was in the centre of the tube here has propagated upward and downward. And the pressure at which this happens allows us to calculate the fracture toughness of that material.

Now, we do these measurements over the full range of temperatures that our pressure tubes are exposed to, in other words, from ambient temperatures, 30 degrees Celsius, up to the operating temperature of 250 degrees and beyond.

So we've done this sort of burst testing on all the pressure tubes that we've removed over the years, but what we're really interested in is how these materials will behave at higher levels of hydrogen than currently exist in the reactors.

So, in effect, testing at higher hydrogen levels allows us to look into the future and predict how the tube properties are going to evolve over time.

So we start by needing to simulate future pressure tube conditions by artificially adding hydrogen to these pressure tube sections from the reactors, and we've developed the means to do so.

So we can now add a starter notch and perform a burst test on one of these pressure tube sections with elevated hydrogen to obtain the fracture toughness data for that particular hydrogen level.

And then from these results, we develop an updated Fracture Toughness Model to account for the influence of this increasing hydrogen. That's what we've been doing over the last several years.

Now, the updated model, as I indicated, covers the entire temperature range from cold conditions through to operating conditions.

And the first important result that we've seen is that there really is no difference in the fracture toughness at normal operating conditions between low hydrogen and high hydrogen levels inside the tube, and normal operating conditions is where the reactor spends the vast majority of its time.

Now, there are changes in fracture toughness at lower temperatures but the reactor only passes through those lower temperature regions now and then when the unit is either starting up or shutting down.

Now, when we do work like this we often ask external third parties to review what we've done and critique it, and in this case we had two different third-party reviewers, one from the U.S. and the other from Canada, and we asked them to evaluate our work.

And they both came to the same conclusion, and that conclusion was that the updated Fracture Toughness Model that we've developed is adequate for application in fuel channel fitness-for-service assessments.

Now, the third-party reviewers made a number of recommendations for enhancements to the model and we will be addressing those recommendations in the follow-up work that we're doing.

And among other things, the ongoing research that we're carrying out will extend the Fracture Toughness Model further into the future.

To date, we've tested hydrogen levels that are beyond where our reactors will be at the 210,000-hour mark that Gary mentioned in his remarks. In fact, we've tested to levels that are several years beyond 210,000 hours and we intend to extend that range further to confirm that there are significant margins on the 210,000 hours.

So let's look at what the results were in a little bit more details.

So here's the schematic of what we found. So to start, what we're going to be showing is fracture toughness as you go up and down on the chart.

So fracture toughness is along the left here. And temperatures, from low temperatures through to operating temperatures, are left to right. The dark green box is the normal operating temperature regime that the reactor is in.

Now, there were a lot of burst tests done on a number of removed tubes in the last several decades and the data from all these burst tests sort of formed a cloud of data points up in this area and over here. And so the authors of the CSA Standard drew two lines below all the data points and those lines are what was put into the CSA Standard.

Now, what that means is that when we need to do an assessment that uses fracture toughness, for instance when we want to assess a pressure tube, the assessment has to use a fracture toughness from these lines.

And this is conservative because we know that the real pressure tubes have fracture toughnesses that fall above the lines. They're better than the lines. So the assessments that we do provide a conservative because we attribute a poorer fracture toughness to the material than we know that it actually has.

Now, all this work was done with

samples with relatively low hydrogen levels. However, we know that we will reach higher hydrogen levels.

So subsequently, in the last I would say three to four years, we've carried out an additional 17 burst tests on artificially hydrided samples and we've hydrided them to a level two to four times higher than the data that was used to generate that line.

And it's the results from these burst tests that we've used to construct the updated model of fracture toughness. And what we've got is shown on this purple curve that's just come in here.

So the first thing we found is that in the operating regime inside the dark green box the data measured with high levels of hydrogen are pretty much indistinguishable from the data with low levels of hydrogen. So there's essentially no change in the Fracture Toughness Model for the operating regime.

At intermediate temperatures, we found some data points above the CSA lines but we also found some data points in this region here below the CSA line.

And so we constructed the curve that's shown here to encompass all the new data from the tests with higher hydrogen and we used this new bounding curve to provide fracture toughness when we now need to do an assessment of a pressure tube.

And again, that provides us a conservative assessment because we're not crediting the materials with the toughness they actually have, we're using a lower bound value of fracture toughness.

So we've adjusted the Fracture Toughness Model to account for the data that we obtained at future hydrogen levels.

So that was the R&D that we've done and how we've adjusted the Fracture Toughness Model.

Now, I'd like to describe what we do with this information and we'll start with the Operating Envelope.

So the fracture toughness of the material is used to determine the allowable operating parameters for the reactor.

And the way we do this is we assume -- we start by assuming that there's a large flaw in the pressure tube. Now, we know that this flaw isn't there but we assume it as a defence-in-depth measure.

So we assume it's a large flaw. It's an inch long, longer than we would ever expect to find. It's almost through a wall but it's not quite through a wall. So it's not leaking, so we don't know it's there. We assume it's there somewhere but we can't be sure that it's there because it's not leaking.

We then calculate the pressure and temperature that the reactor can be at so that this flaw remains stable and then we add an additional margin of 30 percent on the stability calculation and this calculation defines the limits of pressure and temperature that the reactor is allowed to be within. And so fracture toughness has an impact on those limits.

And these limits need to be met for all conditions, normal operating conditions, heat up, cool down, cold shutdown. The reactor always has to stay within those pressure and temperature limits.

The updated Fracture Toughness Model produces a more restrictive envelope in the heat up and cool down range, because that's where the fracture toughness is changed, than the current CSA Model.

And because the fracture toughness value that's used in the assessment is lower, the allowable pressure that the reactor can be at is also lower and so it puts less stress on the pressure tubes. So the change is in a conservative direction.

So that's the first element of defence-in-depth.

Now, let's go on to talk about the second element, which is the need to maintain leak-before-break.

So, as Gary said earlier, the first line of defence in managing pressure tubes is the avoidance of flaws and that's been successful as service experience has shown that we've had no through-wall flaws in any of our operating pressure tubes.

However, as a defence-in-depth measure, we need to demonstrate leak-before-break. So what's leak-before-break?

What we postulate in this case is that there is a similar large flaw almost an inch in length but in this case it starts to leak. It goes through a wall, it starts to leak and it starts to grow in length.

And what we need to be able to show is that we have the ability to detect that the leak is there and that we have clear operator response instructions to the indication that there's a leak and that those operator responses will put the reactor in a safe state, cold and at low pressure, before the flaw has a chance to grow to an unacceptable size. That's leak-before-break.

And the CSA Code requires us to demonstrate that we have leak before break, again, for all reactor conditions, operating, shutting down, in cold shutdown and warming back up.

Now, the authors of the CSA standard explicitly wrote into the standard that you could use probabilistic or deterministic methods to demonstrate leak before break.

Now, probabilistic methods take more data than a deterministic analysis does and we now have that required data. So what we have done is developed the probabilistic method for doing the leak-before-break analysis. And we're adopting the probabilistic evaluation method because it gives a more realistic picture of how the core will behave, and it provides some very useful risk insights.

All right. Let's continue on. So that's the two defense in depth measures. Let's continue on to the last element, which is sort of the overall purpose of fuel channel fitness for service and that's to confirm that there's a low risk of pressure tube failure.

So the -- so per the CSA standard, the whole core has to be assessed - thank you - to demonstrate this acceptably low risk of tube failure, and this assessment is done probabilistically and it has been for a number of years.

Now, core assessments include all the pressure tubes in the reactor, and you don't assess the core as it is today, you assess the core as it's going to be in the future over the next operating interval. So as part of the assessment we have to project the conditions that will exist in future.

Now, probabilistic core assessments have been completed for all our units and the important conclusions are that the current assessments have shown large safety margins compared to the failure frequency requirements that are in the CSA standard.

And the second conclusion is that the updated fracture toughness model really hasn't had a very significant effect on the conclusions of the core assessment. And the reason for this, why this is the case is shown on the next slide.

That is, if you look at the schematic of the pressure tube again, the areas where we expect and where we find higher hydrogen levels are actually quite small regions near the ends of the pressure tube, the areas enclosed in those blue circles. The largest part of the pressure tube remains at relatively lower hydrogen levels, and the outlet end of the pressure tube, where we expect the hydrogen levels to be typically at their highest in that -- in those outlet regions, there are really no fret marks. And if there's no fret marks, there's nothing to grow. And so the result of that is that changing the fracture toughness at higher hydrogen really has very little impact on the overall failure frequency of the pressure tubes.

Now, the -- finally, the requirement that the plants need to meet for failure frequency has been in CSA for a long time. Now, the industry recognized that methodologies have evolved over time and so the requirements are being updated. A study was commissioned to re-evaluate the acceptance criteria and this study has been completed, and it turns out that the recommendations of the study are quite consistent with the CNSC guidance for frequency of Design Basis Events for new plants [RD-337] and our plants meet this recommended value.

Now, as this work is intended to be embedded in the CSA standard, there's a CSA Technical Committee that's reviewing the results of the study, reviewing the criteria, and will recommend changes as appropriate and these changes will be incorporated into the standard by the normal CSA process.

So, in conclusion, we have a well-developed framework of standards that provides an effective means of assessing the condition of our fuel channels.

We have proactively conducted R&D to predict how the fracture toughness of our pressure tubes will change over time.

And we have updated the fracture toughness models to cover these future conditions of higher hydrogen levels.

And finally, we have adopted a probabilistic approach to leak before break as envisaged in the Code to provide a better representation of risk and to provide assurance that we have leak before break.

And finally, I'd like to end with a few important points. The life our pressure tubes is defined by fitness for service. We're confident that our pressure tubes are fit for service and will continue to be fit for service.

We've done tests up to high levels of hydrogen and found no impact at operating conditions where the reactors spend most of their time.

We're doing additional burst tests and we'll be updating the CSA standard with the results of those burst tests.

So the basis -- and the basis for our confidence is the extensive program of inspections that we carry out to monitor the condition of our pressure tubes, and the conservative models and assessment methods that we use to assess the inspection results.

And it's the sum of all these multiple barriers that ensures that our units remain safe to operate.

And with that I thank you for your attention, and Gary and I would be pleased to answer any questions that you might have.

THE PRESIDENT: Thank you. Before getting to the question, I'd like to hear from CNSC staff. I understand that Dr. Rzentkowski has some comments to make. Please proceed.

CMD 14-M15

Oral presentation by CNSC staff

DR. RZENTKOWSKI: Thank you very much, Mr. President. I don't have comments yet. Instead, we prepared a very short presentation to outline our regulatory approach to fitness for service of pressure tubes.

So can we put it up? Thank you very much.

In addition to an update on pressure tube fitness for service presented by the industry, this presentation is intended to provide the Commission with a brief overview of regulatory oversight of pressure tube operation, focusing on regulatory requirements and their effective implementation so that the plant operates in accordance with the licensing basis to assure safety.

From regulatory perspective, it is particularly important that the licensees are managing the actual and future conditions of all systems, structures and components. This implies that the appropriate engineering methodologies, inspections and maintenance programs are implemented to demonstrate future fitness for service to ensure the continued safe operation of reactors.

This also implies that as part of the safety assessment single failure of process systems, such as heat transport system piping, including pressure tubes, are considered.

Also considered are dual failure of a process system and a safety system. For example, a pipe break and a failure to activate the emergency cooling injection.

CNSC staff have a clear and robust

regulatory framework in place to ensure the continued safe operation of reactors. As required by the license condition 7.1, the licensees must implement and maintain programs to ensure fitness for service of the pressure tubes. These programs include engineering capabilities to assess the structural integrity of the pressure tubes and insitu inspections.

The License Conditions Handbook summarizes compliance verification criteria that CNSC staff use in their inspections and assessments of pressure tube fitness for service.

I would also like to add here that the safety assessment of an unlikely event of a pressure tube rupture to demonstrate that the reactor can be shut down, depressurized, and cooled in a timely manner to avoid fuel failures and releases of radioactivity from the plant is performed in accordance with the license condition 5.1.

The specific compliance verification criteria used by CNSC staff are derived from the above standards and regulatory documents. The CSA standards N285.4 and N285.8 provide the licensee with rules to follow for assessing the condition of the pressure tubes as it was described in the presentation provided by the Commission -- by the industry.

The regulatory document RD-334 on Aging Management for Nuclear Power Plants sets out the requirements for managing the aging of structures, systems and components at the nuclear power plant.

CNSC staff activities, including desktop reviews, inspections, and daily surveillance, ensure compliance with all regulatory requirements.

CNSC staff carry out regular oversight of fitness for service of all components important to safety. CNSC staff specifically review and inspect the following outputs from the licensees' programs: life cycle management plans, inspection results, and control room procedures and protocols. This gives staff assurance that the licensees are managing the changes in the physical state of nuclear power plants that occur with time and use.

It is important to note here that ongoing research efforts are directed at refining engineering methodologies and models to conservatively assess the fitness for service of the pressure tubes, focusing on pressure tube leak before break and fracture protection. Leak before break ensures that the reactor can be shut down in a timely manner if a crack develops in a pressure tube, while fracture protection ensures that hydrolysing the pressure tubes have not increased to a level which can render the pressure tubes too brittle to prevent cracks.

Much of the research is focusing on enhancing understanding of physical basis for the linkage between hydride formation and its effect on fracture toughness. Further, burst testing of irradiated pressure tubes is planned to validate the limits of pressure which is allowed in the primary heat transport system at any particular temperature.

The integrity of the pressure tubes is continuously monitored during normal operation by the control room staff. The monitoring includes both, leak before break detection of pressure tubes and break detection in the heat transport system.

In an unlikely event of a leak from a primary transport, from a pressure tube, the consequential leak would be detected in time to shut down the reactor and cool and depressurize the primary heat transport system before the pressure tube rupture.

There are lifecycle management program which confirms the validity of fitness for service engineering assumptions and assessments.

The program for inspection and maintenance includes the periodic inspection program as required by CSA standard N285.4 and fuel general lifecycle management plan as required by CNSC regulatory document RD-334.

Recently the industry has refined the inspection maintenance programs to ensure continued validation of the engineering assessments, including monitoring of the most likely meeting parameters such as the equivalent hydrogen concentration in pressure tubes.

Furthermore, as part of the continued improvement of the engineering models, the industry has committed to a plan for continuing research and development work program on pressure tube aging effects. The purpose of the program is not to validate the current results, but rather to better understand and quantify uncertainties in the models and restriction on the reactor operating conditions where the models are applicable.

In conclusion, I would like to emphasize that CNSC staff ensures that there are inspection programs and lifecycle management programs in place which confirm the validity of fitness for service of all major components, including the pressure tubes.

Furthermore, CNSC staff ensures that the licensees update their inspection program and lifecycle management programs with the latest results from inspections and research. CNSC site staff also routinely inspect the control room to confirm that operators are following approved procedures to confirm that the reactor can be shut down and brought to a safe state in the unlikely event of a pressure tube leak.

As a result, CNSC staff are confident that the industry implemented a proper engineering methodology and inspection and maintenance programs to demonstrate fuel channel fitness for service beyond 210,000 effective full power operation.

CNSC staff will continue to

monitor the industry activities to ensure the continued safe operation of the power plant.

This concludes my presentation which is like a set of comments on the presentation made by the industry. Thank you very much.

THE PRESIDENT: Thank you. I would like to open the floor for questions and I would like to start with Ms Velshi.

MEMBER VELSHI: Thank you. First question to industry, a quick question. You start your presentation that the pressure tubes are safe for the foreseeable future. What does "foreseeable" mean to you?

MR. SAUNDERS: I will get Mr. Newman to answer that.

MR. NEWMAN: It really is more related to the hydrogen concentration, as Dr. Spekkens indicated. So for example, he indicated that around 30 ppm is kind of -- 40 ppm would be what you would expect to see in an as-received pressure tube, we have tested up to 125 ppm, which is a number of years out from where we are at now and it really depends on the reactor and the temperature and the specific conditions of the reactor just how many calendar years that would mean, but you could do those calculations on a reactor-specific basis, but it's a number of years.

MEMBER VELSHI: So with all this testing and modelling that you have done, does the concept of equivalent full power hours sort of fall by the wayside now, that that is not quite life limiting, that you have these more objective quantifiable ways to check how your pressure tubes are doing? I'm just wondering how one relates to the other.

MR. NEWMAN: I think the original designers needed to work from something, so I think it was quite appropriate at the time for them to set a target and then, you know, work towards that.

We, since that time, have been really looking at, you know, really time at temperature because that is really what drives the hydrogen absorption into the pressure tube. And so you are quite right, it is more related to the time that the pressure tube would spend at normal operating conditions where you would be absorbing hydrogen on an ongoing basis. So it really does depend more on that.

In the early life of the pressure tube you will get an irradiated -- like a neutron-induced strengthening of the pressure tube, that happens really in the first five years. After that it basically levels out and doesn't really change dramatically from that point on.

As Dr. Spekkens indicated, it really becomes more of a hydrogen concentration influence when you get out into 10, 15, 20 years of operation.

MEMBER VELSHI: So don't the charts in fact say that if a reactor has been shut down a longer time, the chances of hydrogen build-up are higher than if it had actually been operating at full power, or is it just the heating up and shutting down; it's the up and down as opposed to staying down, okay.

DR. SPEKKENS: No. The ingress of hydrogen to the pressure tubes depends on, as Gary mentioned, time at temperature. When the reactor is shut down there is no hydrogen going into the pressure tubes. So as far as the pressure tube is concerned, time stands still. It's only when the pressure tube is operating at high temperature that the chemical reactions that drive the hydrogen into the pressure tube take place.

MEMBER VELSHI: Can you tell us a little bit --

THE PRESIDENT: Can I -- just for clarification, did you say that on a site-specific reactor, specific site, you will have a new number to replace the 210,000?

MR. SAUNDERS: Yes. I think the answer is there is a new number, but it has not been the effect of full power hours, it's a hydrogen update.

THE PRESIDENT: Well, whatever the parameters are you can come --

MR. SAUNDERS: That's right.

THE PRESIDENT: -- you can come up with a new number that says, okay, so now, with all the experience we have had, with all the research, the new number for this particular plant is X?

MR. SAUNDERS: Yes. We can tell you the number as far as we have done the R&D and as we do more R&D that number may change, it might go farther, but we can tell you that to where we have gone today to about the 124 ppm, as Mr.

Newman mentioned, that that number is safe and how many operating hours that will be will depend on the specific reactor you are looking at.

But we know that number, we do measure the hydrogen in the pressure tubes during our inspection services. So we not only predict it by artificially inseminating it, as we talked about, we also measure it in the tube so we know at any given time where any particular reactor is in terms of hydrogen uptake.

THE PRESIDENT: Thank you. Ms Velshi...?

MEMBER VELSHI: Can you tell us what has been the experience with pressure tubes? If I look at your slide 9 with the different barriers, you know, how many times has barrier No. 1 and No. 2 failed? How many pressure tube failures have there been? How many leaks have there been before break?

MR. NEWMAN: Gary Newman, for the record. We have only had one pressure tube complete failure and that occurred as a result of a pressured tube Calandria tube contact event back in the early days of Pickering A operation.

We have had leaks of pressure

tubes, but not at power conditions, so we have had -- in the early life of Bruce there were manufacturing defects in one pressure tube, they were trying to find that, and so they had it at a partial pressure and that tube began to leak and so they then located it and replaced it, but it was not an at power.

There was then a whole series of activities undertaken to locate sister tubes from that same ingot and then, of course, all those were inspected and cleared.

So other than the one single event which was a pressure tube/Calandria tube contact induced event, there really hasn't been any significant failures.

MEMBER VELSHI: So all the modelling here, then, is based on your R&D work then and I guess whatever comes out from your inspections. I just wonder what your confidence level is with what you have come up with and maybe shed some light on how many pressure tubes have you actually taken out of core and, you know, get them to fail?

DR. SPEKKENS: For the record, Paul Spekkens. The number of pressure tubes we have taken is probably of the order of 20 or 30 over the full CANDU fleet over a long period of time and it is routine with those pressure tubes to do an extensive set of inspections, including a burst test as at the condition that the pressure tube was in in the reactor.

The tests that we have done with elevated hydrogen, as I mentioned, we have done 17 of them and these tests are fairly involved, we can only do about five a year, that is the capacity that we have, so 17 tests represents the last three to four years of effort.

Because we are testing real material that is in service in reactors we are very confident that the properties that we are measuring in the laboratory are representative of the properties that real pressure tubes would have if they were at those hydrogen levels.

So we are quite confident that the results we are getting are in fact valid for the operating pressure tubes.

--- Pause

THE PRESIDENT: Monsieur Harvey...?

MEMBRE HARVEY : Merci, Monsieur

le Président. When you mentioned that fitness for service measurement of the operating life show pressure tube material properties exceeding original estimates, to what extent? Is it way over or -- because a sentence like this, it's difficult to evaluate the significance of this.

MR. NEWMAN: Gary Newman, for the record. What we mean by that is that when the designers originally started to use this alloy they really didn't have as much of an understanding of the material properties and how in fact the pressure tubes would age over time. They didn't realize, for example, at that time that within five years of the first power that really the radiation-induced damage would basically terminate and you wouldn't really get a whole lot of change after that.

They didn't understand what the rate of pickup would be of hydrogen, and so in building the CSA standard we built acceptance criteria that put margins, safety margins and acceptance criteria that have requirements that follow sort of international practices.

So to those who are familiar with the ASME Codes, similar sort of factors of safety of 3 and 1.5 under various conditions would be applied. So we patterned the contents of the CSA standard after similar practices in the U.S., so you would see us being very compatible with that.

So their pressure vessels, for example, would have to be meeting the same kind of requirements as our pressure tubes, so very consistent with industry practices.

MEMBER HARVEY: Just a moment.

--- Pause

MEMBER HARVEY: You mentioned that you have made the probabilistic study for all the units. Saying that, there are so many tubes in one unit, how can we be assured that all of those tubes have been evaluated?

And when you are monitoring those tubes, how can we be assured that everything is monitored?

MR. NEWMAN: Gary Newman, for the record. So you are quite right, there are -- for example, I talked about the Bruce and Darlington reactors, 480 pressure tubes in there. We inspect a fairly substantial number of those pressure tubes, we also sample for hydrogen isotope concentration in a number of those pressure tubes and then we repeat those inspections for both those. We periodically take a fuel channel out and test it.

What we then do -- and that is why the probabilistic methodology is attractive, is we will build distributions of what each of the input parameters looks like. So you could look at, for example, fret depth or fret length, wall thickness of the pressure tubes, hydrogen concentration and these all become distributed quantities which are then used within a Monte Carlo style assessment methodology. That is again following industry practices.

And when you bring all of these combined effects together it produces the likelihood of pressure tube failure which is then compared against, again, industry standard acceptance criteria. All of this is captured within the CSA standard requirements in N-285.8.

MEMBER HARVEY: Okay. You mentioned that the leaks -- eventual leaks would be detected in the gas by the moisture. What is the accuracy of that and the sensitivity of the gas contained?

DR. SPEKKENS: Paul Spekkens, for

the record. The annulus gas system is maintained under very dry conditions, at a very low dew point and the monitors that are watching the system all the time will detect a small change in dew point quite rapidly.

We would be talking about gram quantities of water in a system that would be detectable by the dew point monitors. So the leak detection really is very, very sensitive.

MEMBER HARVEY: But if it is detected, can you locate the leaks right away?

DR. SPEKKENS: Paul Spekkens, for the record. No, you would not detect the leak location because when you detect a leak the first priority is to bring the reactor to a safe shutdown state and that's what leak-before-break is intended to demonstrate.

Locating a leak is done after the fact when the reactor is safely shut down. That is a fairly involved process because the system -- the way the system is assembled groups of pressure tubes are linked together and so you narrow down where the leak is likely to be and then you go and look at those pressure tubes to find the actual flaw, but all of that is done after the reactor is safely shut down.

The first priority in the event that you have a leak is to get the reactor into a safe state.

MEMBER HARVEY: So the location is not an easy task and it could take a certain period of time?

DR. SPEKKENS: Yes. Paul Spekkens, for the record. Yes, that could take time. Sometimes it's more obvious, but sometimes it's not obvious at all where the leak might be.

MEMBER HARVEY: Okay.

THE PRESIDENT: Mr. Tolgyesi...?

MEMBRE TOLGYESI : Merci, monsieur le Président. In your presentation on page 4 in answering Mrs. Velshi's question you are saying that fit for service for the foreseeable future is reactor-specific.

Does that mean that it could vary from reactor to reactor on a multi-reactor site, that it could be -- I don't know, if you have seven or eight reactors, it could be different from reactor No. 1 to No. 3 and No. 5 and No. 7?

MR. NEWMAN: Gary Newman, for the record. Although we said it's reactor-specific, it really doesn't change that much for reactors of a similar number of hot operating hours.

As we indicated, Dr. Spekkens mentioned that it's really the time at operating temperature that drives the hydrogen pickup. And what differs a little bit between reactor designs would be the temperature that the pressure tubes operate at, and so that would have more of an effect than anything else.

MEMBER TOLGYESI: And to staff, how could we regulate reactor-specific base, because if we say that it could vary, could we include that somewhere in the Regulations?

DR. RZENTKOWSKI: That's a very good question, but before I answer the question I would like to go to the previous question which referred to the original design assumptions.

So the specific question was, are we exceeding the specific design assumption. To respond to this question I would like to direct your attention to the slide 15 which shows the fracture toughness of pressure tubes versus the temperature -- that's the industry presentation, sorry.

And you can see here that at

normal operation we are not exceeding the original design requirements of the pressure tubes. They are still valid, we are even slightly above the original design assumptions.

The problem, or the potential problem, may be only in the transition region during the reactor heat up and cool down. And here we have to make sure that the reactor will stay inside the analyzed safety case. And also some of the operating procedures have to be revised -- or has been revised already -- for example, not to use the power heat -- the reactor heat to heat up the primary heat transport system, but simply to use the heat transport system pump to heat up the system from cold to hot.

So basically that means that the reactor will stay critical and we are not going to operate at power during the heat up of the reactor.

So there is a number of considerations we have to factor in our regulatory oversight. One step will be to revise the CSA standards to better reflect the new condition for operation of the reactors. The second step would be to look at the industry procedures in the main control room to make sure that all the conditions are continuously monitored through the main control room. Also we probably would have to look at our regulatory documents, RD-334, which requests the lifecycle management program for critical components and possibly some revision would be required there.

THE PRESIDENT: This is pressure tube 101, so I'm allowed to ask dumb questions I hope.

So at least to me it is counterintuitive, why the toughness goes down at low temperature? It seems to me that everything I am familiar with, the higher the temperature, the worst situation.

Here it seems -- what is the chemistry of things, I don't understand here, please?

DR. RZENTKOWSKI: The system when cold is very brittle, when it is pressurized and hot it is more elastic. That is the reason.

THE PRESIDENT: Do you guys want to elaborate on this a bit?

DR. SPEKKENS: For the record, Paul Spekkens. In fact, most materials do behave the way that the -- or the most metallic materials behave the way that is shown on the slide.

So for example, in iron-based systems there is such a thing as a nil ductility temperature, that if you cool below that temperature the material essentially has no ductility and it becomes very brittle. That is very common in piping systems, in pressurized water reactor, pressure vessels. It is a concern that you can't let the temperature get too low because the material behaves in a more brittle fashion as the temperature goes down.

So zirconium materials are exactly the same way, the values of fracture toughness are different, but the shape of the curve as the temperature varies is pretty similar, and most metallic materials behave that way.

I wouldn't try and explain the chemistry of it to you because I wouldn't be able to, but it is certainly very consistent with what we see in metallic systems generally.

THE PRESIDENT: Okay. That's very helpful, thank you. Mr. Tolgyesi...?

MEMBER TOLGYESI: I see. So when you go these curves, could we consider these curves as the failure frequency curve? That means it's a limit, we should not overpass or be lower or higher than this curve for the toughness?

DR. SPEKKENS: Paul Spekkens, for the record. These curves don't represent a frequency, they simply represent a material property and in the case of specifically, you know, page 15 of the presentation, as the green arrow indicates on the left side, being higher on the graph means the material is better.

So what we consider these curves to be is the curve that bounds all the information we have about all the materials that we have tested and so we use the values from the curves when we do our assessments because we know that real life will be better than that assessment because real life -- in real life the properties are better than the value we are using.

And you could consider it to be a limit; if we found any fracture toughness values below the curve, it would mean that we would have to revise the curve again. We don't expect that. We have done enough tests that we are confident that these curves do bound the level of fracture toughness that we are going to see at the hydrogen levels that we are going to test, but they do represent a limit, but it is a limit in a material property, it is not a frequency of failure.

MEMBER TOLGYESI: You were saying that you did 17 over operating pressure tests and these curves, do they apply the same way to titanium and zirconium pressure tubes or is there a difference, because 17 is not too many.

DR. SPEKKENS: Paul Spekkens, for the record. These curves only apply to the specific material that we tested, which is zirconium 2 1/2 percent niobium.

We don't make pressure tubes out of titanium. Gary's reference to titanium was just that that's a fairly familiar alloy and it is chemically very similar to zirconium and they are both very highly corrosion resistant, but pressure tubes are only made of zirconium alloys.

The tests that we have done here and the curve only applies to the specific alloy that we have out in the field, in the fleet, and that is zirconium 2 1/2 percent niobium.

If there were pressure tubes built from a different material, the work would have to be repeated because the fracture toughness curve is specific to the particular alloy.

MEMBER TOLGYESI: When you were talking about these burst tests you were saying that cannot burst test a pressure tube section with an alloy with an operating pressure range, but could it be over or above the operating range?

So if so, why don't you test at what pressure the pressure tube is bursting?

DR. SPEKKENS: Paul Spekkens, for the record. So in round numbers the operating range for a pressure tube in a reactor is about 10 megapascals, that's the pressure that the reactor operates at.

If we did a burst test without a starter notch we would probably have to pressurize the tube section to about 30 megapascals, to burst at 30 megapascals is a very high pressure.

So to calculate the fracture toughness of the material, the way we do it, fracture toughness is resistance to crack propagation, so to test the fracture toughness there has to be a crack there. That is what the starter notch is, it simulates a crack, then you can burst the tube between zero and 10 megapascals is where these tubes will -- the flaw will start to propagate. If there was no flaw we would have to go to a very, very high pressure, much, much higher than the reactor would ever operate and much higher than the test equipment is able to go to.

MEMBER TOLGYESI: And my last one is, these are destructive testing methods. Do you have any non-destructive testing methods and how do they correlate with these destructive testing methods?

MR. NEWMAN: Yes, Gary Newman, for the record. The only non-destructive techniques that we use would be for actual inspection of the pressure tubes and actually extracting small specimens to test for the hydrogen concentration. Those are all considered non-destructive.

There really isn't any equivalent non-destructive, for example, measuring fracture toughness in situ in a reactor.

Hopefully that answers the question.

THE PRESIDENT: Okay. Dr. McEwan, please.

MEMBER McEWAN: Thank you, Mr. President. Again, if I can go back to try and understand 15, the curves on slide 15.

Very simplistically, if I understood, the risk of a fracture is a function of fracture toughness plus the hydrogen concentration, plus the length of time of exposure to the hydrogen concentration, simplistically.

DR. SPEKKENS: It's not additive, it's a sequential process. So the length of time that you operate a temperature, at high temperature determines how much hydrogen there is. The amount of hydrogen there is determines how much the fracture toughness changes.

At high temperature, in the dark green box, high hydrogen doesn't change the fracture toughness; at low temperature, the presence of the hydrogen is what causes the fracture toughness to decrease.

MEMBER McEWAN: So, in that case, is the time spent in the low temperature region cumulative, i.e., if you spend a lot of time in the low temperature region over a period of years is there a cumulative effect on loss of fracture toughness at the higher temperatures?

DR. SPEKKENS: Paul Spekkens, for the record. No, we don't spend a lot of time in the low temperature region and the chemical processes that drive hydrogen into the pressure tubes are very, very temperature dependent, so those chemical processes would not happen to any significant extent no matter how long you spent in those intermediate temperature regions.

MEMBER McEWAN: Okay. Thank you.

THE PRESIDENT: I'm sorry, just to -- I'm not sure I -- you guys are too fast for me here. I'm not sure, because I thought I heard that the dilemma is when you turn it on and off, on and off, so over 30 years I assume there may have been an on and off. So I thought the question was every time you are in the low and trying to get up you are in the danger of toughness area of brittleness I guess.

So the question is, if you do it many, many times is it a cumulative weakening of the toughness?

MR. NEWMAN: Gary Newman, for the record. It's actually a very good question because fatigue actually works the way that you describe. In this case, though, there is no memory effect, if that makes sense, so the material doesn't really care how many times you have cooled it down and depressurized it, heated it up, it doesn't affect this particular aspect of the material performance.

THE PRESIDENT: Thank you. Anybody? Monsieur Harvey...?

MEMBER HARVEY: Just one question. On slide 14, when you say the third-party reviewers, it was two experts I think. When you hired those experts do you talk with CNSC or does CNSC have to accept those experts?

DR. SPEKKENS: Yes, Paul Spekkens, for the record. Yes, the third-party reviewers are agreed to with the CNSC staff to make sure that the reviewers are acceptable to the staff.

THE PRESIDENT: Anybody else? I just have one -- I have maybe two questions here.

The first one is maybe to both you and CNSC. It seems to me in listening to this that we are assuming that hydrogen and toughness here of the material is really the things, the proxy for aging of the tube. What I am always -- particularly in the nuclear, what I am always worried about is the known/unknown or the unknown/unknown.

Is there any other possible kind

of a phenomenon that would impact the aging of the pressure tubes that we should worry about that is not in your model here?

MR. NEWMAN: Gary Newman, for the record. So one of the reasons that we periodically remove fuel channels and test them, put them -- as Dr. Spekkens mentioned, put them through a battery of tests is to confirm that in fact other aspects like fatigue or other degradation mechanisms are not present. So we are looking for the unknown, if that makes sense.

And then we have also used that material to accelerate the age of the pressure tubes through the doping mechanism for hydrogen to try and look out in time.

THE PRESIDENT: So, CNSC, when you will have to make judgment on a site-specific -- and I am now maybe alluding to the upcoming May hearing -- what is it you will need in terms of actual data to decide about the Pickering situation, for example?

DR. RZENTKOWSKI: There is more aging-related degradation mechanisms which affect the fitness for service of the pressure tubes.

The second important one is

blister formation on the surface of the pressure tube as a result of a contact between pressure and Calandria tubes. This particular aging mechanism appears to be quite important for Darlington reactors.

The one discussed here, which is the hydrogen intake, is predominant in other reactors, so that means Pickering and Bruce reactors, and that is why the focus of this presentation was on hydrogen intake, because it's directly related to the fractural toughness of the pressure tubes.

But there are other degradation mechanisms which we are monitoring and a recommendation will be site-specific if it comes to the relicensing decisions.

THE PRESIDENT: Well, what do you mean "if"? In May I thought there is a known announced meeting to deal with the site-specific.

DR. RZENTKOWSKI: I'm sorry, I meant to say "when", not "if". Thank you.

THE PRESIDENT: Thank you. Anybody else? Does anybody want to add anything else, a final word?

Okay. Well, thank you very much.

I found it very educational, so thank you for that.

--- Pause

THE PRESIDENT: Okay. We will resume after lunch at 1:15. A short lunch, thanks.

--- Upon recessing at 12:24 p.m. / Suspension à 12 h 24

--- Upon resuming at 1:20 p.m. / Reprise à 13 h 20

5.3 Presentations on Probabilistic Safety Assessment (PSA)

THE PRESIDENT: Okay, I think we are ready to proceed to the next item on the Agenda, which is a Presentation on the Probabilistic Safety Assessment.

I understand that there is going to be again a joint presentation by OPG, Bruce Power and NB Power as outlined in CMD 14-M16.1.

I understand, Ms Swami, you will make the presentation.

Please proceed.

CMD 14-M16.1

Oral presentation by Bruce Power, Ontario Power Generation Inc. and NB Power

MS SWAMI: Thank you, President Binder and Members of the Commission. For the record, my name is Laurie Swami, Vice President Nuclear Services for OPG.

Today the Canadian utilities are here to provide you with an overview of the probabilistic safety assessments that we use as part of the overall safety assessment at our respective facilities.

This presentation will provide you with an overview, while specific facility presentations will be provided when each of the utilities is before you in future. Our presentation today will be provided by Dr. Jack Vecchiarelli and Mr. Frank Saunders. Dr. Vecchiarelli is the Nuclear Safety and Technology Manager for OPG. Mr. Saunders is the Vice President of Regulatory Affairs and Nuclear Oversight for Bruce Power. In addition, Mr. Derek Mullen, Senior Technical Advisor Reactor Safety from Point Lepreau is here to answer any questions you may have.

Dr. Vecchiarelli...?

DR. VECCHIARELLI: Thank you, Laurie. For the record, Jack Vecchiarelli, Ontario Power Generation, Manager of the Nuclear Safety and Technology Department.

On behalf of the industry I would like to give you a presentation on the topic of probabilistic safety assessment or PSA. The main purpose of this presentation is to provide an overview of PSA, in particular, how it is used as a tool to assess risk. The presentation will cover a number of aspects, including the concept of risk and safety goals, an overview of PSA methodology, the various uses of PSA, various PSA improvements to methodology, a perspective from the whole site approach to risk assessment and a summary at the end.

The word "risk" is commonly used in everyday language in one context or another. Risk can be used to indicate the degree of safety of an activity, recognizing that there are inherent risks in all human activities such as driving a car. In general, risk is the likelihood of an undesirable event multiplied by the consequence. It provides a common basis for comparing the safety of different types of activities.

For example, this chart shows how risks can be qualitatively characterized in terms of high, medium, low risks, depending on the likelihood and consequences of events. For instance, on the top right corner of the chart we would say that the risk is very high if an event is almost certain to occur and the consequences are extreme. In contrast, on the bottom left corner of the chart we would say that the risk is low or very low if an event is rare and the consequences should the event occur are negligible.

With respect to nuclear power plants, risk is assessed both qualitatively and quantitatively. When considering risk the fundamental question that must be addressed is, "How safe is safe enough?" Recall that there are inherent risks in all human activities; as such, the risk is not zero.

So for nuclear power plants we need to decide what very low level of risk are we willing to tolerate. The criteria that delineate between tolerable and intolerable risks are often the most challenging to determine. This is where the development of safety goals is important. Safety goals are established to help us address the question of how safe is safe enough. They are used to guide us in determining when improvements or corrective actions should be implemented to reduce risk.

That said, we do not base nuclear safety on just one safety goal or a single numerical value. Conceptually there is a hierarchical set of qualitative and quantitative safety goals. This is consistent with the emerging view of the International Atomic Energy Agency. At the high level there is a qualitative goal which sets the overarching societal objectives that are deemed to be acceptable, while at the lower level there are very specific quantitative goals which include criteria that are based on safety analyses from both deterministic and probabilistic assessments.

This is a simplified depiction of the hierarchical safety goals framework to illustrate the concept. At the top level we have qualitative goals expressed in terms of the protection of the public. This would relate to health objectives such as limiting the probability of fatalities due to radiological releases from nuclear accidents.

In the intermediate level we have semi-qualitative goals that are performance-based or action-oriented. These include the various programmatic activities that the licensee carries out, such as equipment testing and maintenance; staff training and the fostering of a nuclear safety culture; emergency response, including the use of severe accident management guidelines. These are all activities that the licensees perform on-site. They are actually doing something.

As we progress down this pyramid, the goals become increasingly more quantitative in nature. At the low level are the specific quantitative goals and criteria that include those associated with the safety analysis and PSA. It is through all of these supporting elements collectively that nuclear safety is assured and the high-level health objective is met. The foundation for safety is built upon defence in-depth principles in the achievement of the specific safety goals.

Here are some examples of how the high-level qualitative safety goals are defined. In Canada, the *Nuclear Safety Control Act* stipulates the prevention of unreasonable risk to the environment and to the health and safety of the public. The IAEA fundamental safety principle is articulated in terms of protecting people and the environment from the harmful effects of radiation. The U.S. NRC has two qualitative safety goals. One states that individuals should bear no significant additional risk to life and health. The other states that nuclear power plant operation should not be a significant addition to other societal risks. Note that these are expressed in terms of relative risks, that is compared to other risks that the public is normally exposed to.

To help implement and meet the high-level safety goals, the Canadian nuclear industry has developed and used quantitative PSA safety goals. These safety goals are more stringent than the health objectives which are associated with the high-level qualitative goal.

It has been shown that by meeting these so-called surrogate safety goals the overarching health objectives are met.

The industry safety goals are expressed in terms of limiting the frequency of occurrence of certain undesired consequences. For instance, as shown on the slide, we have a safety goal for severe core damage frequency expressed as the number of times per year that a severe core damage accident may occur for your reactor unit. Of course, we wouldn't want this to happen even once per year so we have a goal that is in terms of a very small fraction of one; namely, 1:10,000. Basically the severe core damage frequency represents the probability of a severe core damage accident occurring in the next year. But in the unlikely event of a severe core damage accident, a large release outside of containment may not necessarily occur, so the risk of a radiological health effect on the public is much lower.

Similarly, we also have a safety goal for large release frequency expressed as the number of times per year that a severe accident with a large release to the environment may occur for a reactor unit. Basically the large release frequency represents the probability of a large release accident occurring in the next year, with the goal being 1:100,000. In such an unlikely event, other factors come into play and the risk of a radiological health effect on the public is even lower still.

From an international perspective, the IAEA supports the same type of safety goals and numerical values for operating plants, 1:10,000 for core damage frequency, 1:100,000 for large release frequency.

In addition to the same pair of quantitative safety goals for operating plants, the U.S. NRC also has two semi-quantitative health objectives. The first is in terms of a relative risk of prompt fatalities being .1 percent of that from other non-nuclear accidents that the public is normally exposed to. This equates to approximately 1:2 million per year. The second is in terms of a relative risk of cancer fatalities, being .1 percent of that from all other causes. This equates to approximately 1:500,000 per year.

The Canadian nuclear industry has considered similar health objectives to justify the severe core damage frequency and large release frequency safety goals.

In summary, the industries per reactor per year safety goals are consistent with the IAEA and U.S. NRC safety goals for operating plants. Furthermore, the Canadian industry has treated the quantitative safety goals as limits and has adopted target values which are one order of magnitude lower. The targets and limits are used by industry to manage risks as part of a risk-informed approach. I will elaborate on this later in my presentation.

In relation to the concepts of risk and safety goals, probabilistic safety assessments have been developed as an analytical tool to, one, improve the plant design and operation and maintenance practices; and, two, to quantify risk for comparison with safety goals.

Basically PSA answers three questions: What might go wrong; in other words, what are the events and possible sequences of events; what are the potential consequences of these event sequences; and, what is the likelihood that these events sequences may actually occur?

In a systematic manner a PSA numerically assesses the probability and consequence of foreseeable events and evaluates the risk for each. This slide shows the three types or levels of PSA: Level 1, 2 and 3.

In a Level 1 PSA, various initiating events are taken as input and the responses of the plant systems are modelled considering the different possible combinations of failure modes that could lead to different possible event sequences and outcomes that are expressed in terms of damage to the fuel in the core. The output from the Level 1 PSA is an estimate of the severe core damage frequency.

In a Level 2 PSA, the results from the Level 1 PSA are taken as input, the reactor core and containment behaviour is modelled and this considers different possible containment impairments and failures which lead to different possible outcomes that are expressed in terms of releases of radioactivity to the environment. The output from the Level 2 PSA is an estimate of the large release frequency.

Now, at this point there is sufficient information to evaluate the industry surrogate safety goals and, therefore, support that the health objectives are met.

Although not necessary, a Level 3 PSA takes the radiological releases to the environment and estimates the off-site consequences, for example, in terms of public doses. This takes into account weather conditions, protective measures, land contamination and other factors. The output from a Level 3 PSA includes an estimate of the public health risk. However, the degree of uncertainty in the PSA results increase substantially for a Level 3 PSA compared to a Level 1 or 2 PSA. For this reason Level 3 PSAs are not usually performed. The costs of performing PSAs also increase as the scope moves from a Level 1 to a Level 2 to a Level 3.

Now, I have already covered most of the information in this slide, however, I would like to point out a few things here. Level 1 and Level 2 PSAs are required in order to meet CNSC Standard S-294. Also, the PSA results for severe core damage frequency and large release frequency are expressed on a per unit basis, that is, the focus is on what happens to a single reactor unit. However, multi-unit effects are considered in this process.

For example, in a common mode event such as a station blackout where all units are simultaneously affected by the same initiating event, the total release from all units is considered in the large release frequency calculation. One must bear in mind that the LRF calculation is a summation of the frequencies of occurrence of different possible events that each result in a large off-site release, so the per unit LRF actually captures a mixture of contributions from purely single unit events, as well as multi-unit events due to common mode scenarios.

Lastly, the PSAs include both internal and external events, that is, events that originate inside the plant, such as LOCOs, as well as those originating outside the plant such as high winds.

There are separate PSAs for each of these different types of hazards; a PSA for internal events, a PSA for seismic, fire, flood, et cetera.

One of the reasons for having separate PSAs is that for any given PSA we are looking for insights on reactor safety. Different insights may be gained from each of the different PSAs depending on the type of hazard analyzed; hence, it is important to look at them separately.

This figure is meant to highlight that there has been a continual development of PSA by the industry over many years. As early as the 1980s the industry has been very active and has worked collaboratively in undertaking substantial work to conduct station PSAs and PSA updates. The PSAs are very high quality and are consistent with best industry practices. In some cases the Canadian industry has been at the forefront of PSA methodology development.

Prior to S-294 the industry focus was on Level 1 and Level 2 PSAs for internal events. Valuable risk insights about internal events were gained during that time. As this PSA methodology matured and S-294 was phased in, the focus grew to include Level 1 and Level 2 PSAs for external events as well as internal events.

The PSA regulatory requirements in S-294 proved to be comprehensive and prudent, and additional risk insights have been gained as part of S-294 compliance efforts and the inclusion of external events in the scope of PSA.

Attention has now focused to whole-site PSA, where Canada is leading the way internationally on pioneering a methodology. We hope that with -- as with past PSA developments, further risk insights will be obtained by whole-site PSA.

PSA models are quite large and complex. To give you an idea, in a typical PSA model there are over 30 distinct initiating events that are modeled, such as loss of coolant accidents, stream line breaks, loss of electrical power, loss of service water, et cetera.

Over 30 plant systems are represented, shutdown systems, emergency coolant, injection systems, emergency power systems, et cetera.

With tens of thousands of individual potential failures modeled at the detailed equipment and component level, supported by a large database of information gathered from equipment testing and other sources, it is very labour intensive to develop these detailed models. For example, it takes over two years to create a Level 1 PSA model. One needs this level of detail in order to properly determine the probability of the various sequences of events that can lead to severe core damage. Given the robustness and high degree of defence and depth of nuclear power plants, a series of multiple failures has to occur for an accident to progress to severe core damage. To estimate the probability, one needs a detailed model of the various mitigating systems and the backup systems down to the intricate component level of the system equipment, pumps, valves, instrumentation, et cetera.

To give you an analogy, in estimating the probability that automobile brakes fail when you need them, one would need to model the various mechanical components that constitute the braking system and the failure probabilities of each, including the brake pedal, front and rear brake pads, brake fluid system and master cylinder, the emergency brake, the nuts and bolts, et cetera. You can imagine the complexity and much greater amount of detail needed to model an entire nuclear power plant.

It costs in the range of 10 to $50 million to complete a PSA. The total industry spend is roughly a hundred million dollars for the suite of PSAs, where each site has 10 to 15 separate PSAs, including internal events and external events, such as seismic, fire, flood, high wind, et cetera.

One of the most important uses of PSA by the industry is for risk management. Insights gained from PSA are used to identify potential improvements in station design and operation. PSA is used as input to risk-informed decision making. For example, it is used in support of outage planning, where certain equipment may be out of service due to maintenance and compensatory measures are put in place. Such cases present particular plant configurations that need to be evaluated in terms of tolerable risk levels.

As mentioned earlier, safety goal targets and limits are used by industry to manage risk as part of a risk-informed approach. On average over the course of a year the plant risk is not expected to change significantly. The industry strives to meet the safety goal target values to the extent practicable, while ensuring that the safety goals are within the limit values.

However, should new information come to light indicating that a safety goal exceeds the limit value, corrective actions would be undertaken to reduce the risk to be below the safety goal limit. This could include a closer examination of any PSA modeling conservatisms or possible changes to the plant operation or the design.

It is important to understand that PSA is more than just about the numbers. The main benefits of PSA are the safety insights gained from the process itself. PSA helps to identify the relative contributions to risk from the different hazards, as shown in this pie chart for illustrative purposes. For example, depending on the station design, steam line breaks have been found to be significant contributors to the severe core damage frequency. Other contributors include small loss of coolant accidents, as opposed to large loss of coolant accidents for which the relative risk is small. Events involving loss of electrical or water supplies are also important. These insights raise awareness of the importance of particular systems used to mitigate such events. The risk significance of different systems are evaluated to identify systems important to safety, which are given special attention as part of the industry's reliability programs.

In terms of improvements to PSA methodology, as I've already mentioned, the inclusion of external events has been a major part of the revised PSAs by the industry.

On this topic of external events, following the Fukushima accident there was heightened awareness of the importance of external events. In response, industry procured emergency mitigating equipment, EME, such as portable diesel generators and portable pumps, to further improve safety. The EME represents an additional layer of defense in depth that bolsters the robustness of nuclear power plants for the mitigation of highly unlikely beyond design basis events.

The industry is working together to include EME in the PSAs as appropriate. For example, as sensitivity cases to try to quantify the risk/benefit of EME. There are some challenges in attempting to quantify the risk/benefit by means of PSA. However, as part of continual PSA methodology improvement, the industry working its way through these challenges. Early indications are that EME, together with other PSA modeling improvements, can reduce the estimated risks for both internal and external events. Depending on the scenarios, the risk reductions are expected to be approximately a factor of 2:10.

I will now turn over the presentation to Mr. Frank Saunders of Bruce Power to discuss the whole-site perspective to risk assessment.

MR. SAUNDERS: Frank Saunders for the record. So as you've heard through the presentation, we have used a unit-based or unit-focused PSA for 20-plus years in industry with a growing sophistication in our application of that capability. So what then is the whole-site of PSA and what does it mean in relative terms to the single unit PSA? So we're going to take a few slides to try and discuss that with you.

The site-focused PSA really only came into the discussion a year and a bit ago and for a couple of reasons. I mean, one is some interveners have asked the question, well, what's it look like as a whole-site, which is kind of a logical question to ask and to provide an answer to.

And the second is as we have added more capability in our post-Fukushima reviews in terms of the emergency mitigating equipment and our emergency response capabilities, it makes some sense to look then at how those things interact and determine whether there's -- you know, what additional benefits and so forth we could gain.

So from -- I have a little diagram here on the right, which is illustrative but I'll take you through it just to kind of establish what the differences are.

So in the centre, of course, is the deterministic analysis and goals, which were the way these plants were designed and licensed originally. PSAs weren't really a method that was employed at the time. And then over the years we've added the plant-specific, the unit-specific probabilistic safety analysis, which included some of the next string, the external events. And that was, in essence, the makeup of the unit-based PSA material.

So since Fukushima we have expanded the number of external events, gone to, you know, a broader range of events with even lower frequency. So we've greatly expanded that piece, and those expanded is being added into the unit PSAs as we speak. Some people have completed it and some are nearing completion.

The next ring is the emergency and protective equipment, which is, in fact, a whole new layer of defense, which, as you just heard Jack say, is now being added into the unit-based PSAs as well.

Beyond that, you know, when we looked at the unit PSAs, we really considered the unit itself and the resources in the unit. We didn't look at the site-wide resources that were available, and this is -- this is anything from people. You know, if you've got a four-unit station and you've got difficulty with one unit, obviously you've got three other units worth of staff who can be deployed to assist and help in that event. You've got emergency mitigating equipment, emergency response forces, various things from the site that you could use to help mitigate or prevent that event, and those aren't currently used within the single unit PSA analysis.

And SAMG Response, again, just

developed fairly recently, in the last five years or so. Again, looking at a situation where there's some sort of a severe accident underway, how you can either truncate or mitigate that accident by using some rather unique, you know, procedures and equipment available to you. This is somewhat modeled in the unit PSAs, but not to a large degree, some pieces.

And then, of course, emergency management is the final loop. At the end of the day if there is an event and there is some release, emergency management, a. allows you to limit it, and b. make sure that you can get people out of harm's way. So it ultimately affects the impact of the event.

So both types of these PSAs, of course, you know, have a particular perspective which could add value, but they are different in terms of how you look at the numbers and you really do need to think about how you're going to model them because in the PSA the modeling is really the foundation of the analysis. You need to look at that modeling and make sure it's accurate; otherwise, whatever output you get won't be very meaningful.

So on the whole-site perspective, we're not alone in beginning to look at this. It really is at the very infancy from an international practice. Nobody has actually developed a good way of doing this yet. We are certainly, I think, in the forefront. We claim we're leading, but we're at least at the front of the charge and we have already held one workshop, an international workshop, in January in Toronto to look at this and I know CNSC is planning a workshop for November, I think. So I think Canada is actually, you know, out in front of most of the world in considering how you establish and look at a whole-site PSA.

So the first question you want to ask when you're looking at the whole-site PSA is what is it that we're trying to see; right? What can we gain from looking at a whole-site PSA that we wouldn't gain from looking at the unit PSA?

And to do that, you need to consider some effects. And Jack has mentioned common mode events and this is true. Common mode events are already calculated in the unit PSAs. In fact, almost half of the initiating events and unit PSAs are actually common mode events which affect all the units in that particular station, not just a single unit. So certainly there's a significant amount of double-counting in the unit PSA. And that's okay because what you're looking at in the unit PSA is to understand the sequence of events, the failure mode, so you don't really care that you duplicate something that might be affecting another unit, it's just not important in that discussion. You just want to know what affects the unit.

So if you want to look at the whole-site, though, and understand the true comprehension of the risk of the site, you need to separate these out a little bit.

There are certainly different levels of realism and uncertainty in calculations for things like external weather events than there is on equipment failures. You can on equipment failures model system very accurately, you have lots of failure information both from your own plant and from many other plants around the world. So you can make it, in fact, a fairly accurate model of what the probability of failure in any given system or equipment is.

When it comes to external events

and weather events in particular, that level of certainty is just simply not available because just the information and the unpredictability of the weather is large enough that you can't be that certain. So when you create your models, you have very different certainties.

The other question, of course, is the people side of the equation and creating the EME and the response capability to any kind of an event. You know, we've created a very diverse way of approaching it by using portable equipment and people. People are by their nature very flexible. They're the most flexible resource we have. We can count on them to think and move and react to things in a way that a simple piece of equipment cannot do.

On the other hand, people from a reliability point of view are harder to model; right? You know, you can have errors from people, just as you can have failures in equipment, but much harder to predict what that would look like when you're dealing with an operating force, so ... So, again, it's a different kind of analysis, an important analysis, but not quite the same as what you do inside the plant.

So to make sure that we get good clear insights, we must take some care in completing this work to make sure it's good. Simply adding up the per-unit PSAs will not give you an accurate number for sure. We know it will double count a lot of things.

So the question is how might we go about this? Well, some things we can tell you very clearly now. We will consider all radioactive sources. In the original site PSAs we didn't consider anything but the reactor as a radioactive source. So we'll include new fuel pools and fuel storage and anything that's there that has the potential to release activity even though the likelihood and the probability for some of these sources would be much, much smaller than the reactor, of course.

We will separate out the common mode events that are used in the per-unit PSAs and then factor them back in on a station basis rather than on a unit basis. And I can give you another example. You know, Jack talked with an event, but a similar sort of thing is looking at external events as high wind or some of those events. Again, when we looked at that on a per unit basis, we calculated the failure on a per unit. But when we looked at the release that would be necessary to trigger a large release, we considered a station, not a unit. So that unit reflects actually the risk for the whole station in that particular event. So we need to factor those out of the single unit one and factor it back in at the station level. And that takes some work. It's not quite as simple as it sounds. You know, it sounds easier than it is because you've got to run the models is the issue, and you've got to create the models and validate them and verify them.

Different hazard types will be considered, internal and external, and we need to think about how we add those things. You know, to start with, the probabilistic safety analysis indicates that the level of risk is already a very small number, usually in the range of 10 to the minus 5 or 10 to the minus 4, so uncertainties can have a -- you know, a pretty big impact on that small a number, so we do have to be sure we get them reasonably ac-- accurate, excuse me.

And we will consider different operating modes. And we do this in PSAs today, of course. We consider both shutdown and operating when we do PSAs. So we look at the risk in the two modes. Of course when you look on a site basis you may have units that are in different states. So some units might be shut down, some units might be running, and so you need to think from a site basis of what that looks like. And the risk, of course, is different depending on the mode that you're in. So some way of assessing what that looks like also needs to be considered.

So the path forward that we looked at is pretty much what you seen on the screen here. We consider this to be a fairly aggressive schedule actually and I think you can sort of back that up a little bit if you look at the -- that slide 15 and the work we've done over the years. You know, two or three years to develop these models well is pretty much a challenge as it is.

So we will -- you know, we're aiming to hit these targets for '16 and '17. We will, of course, not sort of be silent between now and 2017, we will provide updates and information as we go through the various factors and develop the goals. And of course we do want to stay very close to the international community to make sure that what Canada develops here is not inconsistent with where the rest of the world will ultimately go.

So I'll turn it back to Laurie just for some conclusions.

MS SWAMI: Laurie Swami for the record. In closing, the Canadian utilities have been at the forefront of developing and implementing PSAs since the '80s. We use this tool as one input to managing nuclear safety at our facilities. We use the results to assess the design and operational enhancements, which enable us to continuously make safety improvements.

We are pleased to continue enhancing the PSA framework by developing appropriate methods for determining and assessing site-wide risk values. We have taken a good first step by hosting an international whole-site PSA workshop this past January, where we were honoured to have U.S. NRC commissioner, Dr. Apostolakis, actively participate.

We look forward to participating in the CNSC workshop this coming fall.

We would like to thank you for this opportunity to provide an overview of our PSA programs. We are available to answer any questions you may have.

THE PRESIDENT: Thank you. Before question period I'd like to turn to CNSC staff and I understand, Dr. Rzentkowski, you have the floor.

CMD 14-M16

Oral presentation by CNSC staff

DR. RZENTKOWSKI: Thank you very much, Mr. President. Our presentation focuses on the licensing requirements stemming from the general safety and health objectives and their effective implementation so that the likelihood of severe accidents is extremely low and the potential radiological consequences are limited as far as practicable.

The general regulatory approach is consistent with the currently accepted international practice except the work on aggregation of risk contributors, a multi-unit station which extends beyond the temporal horizon of the present day.

CNSC staff have a clear and robust safety analysis framework in place which consists of deterministic and probabilistic approaches, which are complementary in nature. It has to be understood, however, that the current approach is largely deterministic to clearly demonstrate that the dose acceptance criteria are met and to approximate conditions associated with beyond design basis events.

The PSA, on the other hand, identify risk contributes from postulated initiating events to the total risk to balance strategies for accident prevention and mitigation, and in turn identify potential safety improvements. The CNSC is one of the few international regulators that require full scope Level 1 PSA, which identifies the sequences of events leading to core damage, and Level 2 PSA, which identifies the sequences leading to large releases to the environment.

The gradual approach to compliance with S-294 started in 2010, when the document was referencing the power reactor operating license for the first time. Currently the PSA must be updated on a periodic basis. The dates for the PSA updates are specified in the License Conditions Handbook. The purpose of the revisions is to apply latest models or reflect improvements to the reactor design. The deterministic analysis used in the safety reports also undergoes the same type of periodic updates.

The new regulatory document 2.42, which replaces S-294, is going before the commission later today for approval, of proposed amendments triggered by lessons learned from the Fukushima accident, including requirements for multi-unit station impacts, site-specific external events and their potential combinations, and eradiated fuel bay analysis.

The regulatory requirements pertaining to the application of PSA stem directly from the *Nuclear Safety and Control Act*, which requires prevention of unreasonable risk to the environment and to the health and safety of persons. The design requirements for new power plants, defined in RD-337 Regulatory Document, include qualitative safety objectives and clear safety goals that a reactor design must meet to minimize any significant additional risk to the public in comparison with other risks to which the public is normally exposed.

The safety goals are defined as

the frequency limits of the sum of all event sequences that can lead to:

- large release that would trigger long-term relocation;

- small release that would trigger temporary evacuation; and

- severe core damage as reactor defence-in-depth principle.

The core damage frequency safety goal relates to accident prevention, whereas the release safety goals, which include release limits, relate to accident mitigation.

These safety goals are formulated in addition to the deterministic design requirements and the dose acceptance criteria so that the risk of accidents that originate outside of the design basis of the plant is considered.

The PSA is the analytical instrument used to determine whether the reactors meet the safety goals.

Please note that the new Reg Doc, Regulatory Document 2.5.2, which replaces RD-337, is going before the Commission today for approval of proposed amendments, including requirements for portable emergency mitigating equipment, design extension conditions to account for several accidents, and spent fuel storage.

This slide presents the conceptual basis for the development of the current safety goals.

There are two fundamental health effects on the public, one relating to early fatalities, another relating to late or delayed fatalities.

Early fatalities are typically linked to accident rates such as industrial and traffic, while late fatalities are linked to cancer rates.

These health effects are expressed in terms of the additional risk of a fatality caused by the reactor operation being a very small percentage, typically less than 1, of the background cancer risk.

The table on this slide compares Statistics Canada's data for early and delayed fatality rates and shows that the incremental contribution to the public health risk from a nuclear accident is in fact less than 1 percent.

The actual numerical safety goals are conservative surrogates to these health effects to simplify their calculation. The frequency is defined per year to assure a consistent approach to the reactor design. It is conservatively assumed in the calculation that the average member of the most critical group refers to an individual at the site boundary.

The release safety goals, defined by CNSC staff as the frequency and the release limits, establish the conditions that would trigger evacuation or permanent relocation in consideration of societal effects of nuclear accidents.

This differs from international practice where the criteria are based on the frequency of radiation-induced fatalities but is consistent with lessons learned from the Fukushima accident. Therefore, there is no recommendation made with regard to safety goals in the design requirement document. It has also more practical meaning to the public and it is less subject to interpretation.

It should be noted here that although the frequency limits for safety goals are generally accepted internationally, its application is inconsistent. Countries with large nuclear programs such as France, Canada and the U.S. define them per reactor year, while countries with small nuclear programs such as Finland, Switzerland or Sweden define them per year.

The review of licensees' PSA methodology is the first step in review of PSA as per S-294 document. It includes the acceptability of the proposed probabilistic safety goals proposed by the industry. The methodology is reviewed against the IAEA Guides listed in S-294.

The detailed review of PSA results includes, on a sampling basis, evaluation of the models and verification of final risk contributors to the total risk, representing the core damage and large release frequency goals.

For existing plants, the safety goals are used to estimate the frequency of accidents which potentially may lead to significant radiological consequences. They support the evaluation of design upgrades to reduce the risk of reactor operation to as low as reasonable but are used as expectations rather than mandatory regulatory requirements.

The safety goals are set as the frequency limit and target. The limits should be no more than one order of magnitude higher than that for new power plants, while the target should be the value established for new power plants.

The regulatory approach is as follows:

- if safety goals are above limits, then modifications are necessary;

- if safety goals are between limits and targets, then modifications are necessary if practicable; and

- if safety goals are below targets, then modifications are expected if practicable.

However, the safety goals are not the only basis for regulatory decision as they are considered in addition to design requirements and release limits.

The PSA has originally been developed for a single unit for internal events only. Recently, it has been extended to include external events and will further be extended to multi-unit stations.

However, the simple summation of internal and external events for a unit could provide an extremely conservative estimate. The summation should be performed only once the bias due to uncertainties and conservative assumptions associated with the methodology for external events is removed.

As discussed, safety goals are defined in terms of events per reactor year and therefore the effect of adjacent units are considered and accounted for when calculating safety goals for internal event sequences.

Nevertheless, in terms of events per site per year, the frequency of an accident for certain internal event sequences would be higher at a multi-unit station in comparison to a single-unit station.

On the other hand, for certain external event sequences which may affect all units at the same time, the frequency of an accident per reactor at a single and multi-unit station would be the same.

It follows that one cannot simply multiply the unit safety metrics by the number of units to calculate the safety goals.

To address the problem of risk aggregation, CNSC is leading the international community in development of safety goals with a plan to complete this task by December 2014.

To conclude, I would like to highlight several important points raised in this presentation.

Safety goals are used as numerical surrogates of safety and health objectives. They are formulated in addition to the deterministic design requirements and the dose acceptance criteria.

In principle, implementation of safety goals ensure that the likelihood of accidents with serious radiological consequences is extremely low and that potential radiological consequences from severe accidents are limited as far as practicable. Thus, the safety goals enhance the overall level of safety but they are not the only basis for regulatory decisions.

Safety goals provide essential information for decision-makers with regard to acceptability of new designs and assessment of design upgrades to reduce the risk for operating reactors to as low as practicable.

All operating power reactors in Canada meet or exceed the established safety goals. It is expected that the Fukushima-related improvements will further reduce the calculated safety goals.

The implementation of safety goals in Canada is consistent with currently accepted international practice.

In the spirit of continuous improvement, several initiatives have recently been undertaken by CNSC staff to develop a regulatory position on multi-unit metrics for safety goals.

This concludes my presentation. Thank you very much.

THE PRESIDENT: Okay, thank you. A lot of material to absorb here.

I'd like to open up the questions with Ms Velshi.

MEMBER VELSHI: Thank you, Mr. President.

This is a very complex area and I'll ask for forgiveness right at the outset if the questions are lame and basic and perhaps don't even make sense.

I get very confused when I hear safety goals as being a surrogate of safety and health objectives but I don't see a connection between the two. So, as a lay person, do I really care about large release frequency and all that? I really want to know what's the increased risk of fatality or cancer and I don't see the connection between those.

So, as I look at the U.S. NRC safety goals, I actually see something there, but maybe you can help me understand the linkages between the safety goals established versus these health outcomes. And I'll ask staff to have a first go at it please.

DR. RZENTKOWSKI: And this is what we try to explain on Slide 4, to link the general health objectives to quantitative health objectives and then explain that safety goals are defined as a surrogate of health objectives.

MEMBER VELSHI: Yes, I know the slide is there, I just couldn't understand what that really means. So if I live in a highly populated area for instance, isn't the risk of a health outcome higher than if the plant wasn't --

I see the numbers. I just don't know what that really means when I look at what that means as far as health outcomes.

THE PRESIDENT: Can I add my

frustration to this?

DR. RZENTKOWSKI: Yes.

THE PRESIDENT: You guys keep talking about qualitative. Yet, I see a lot of quantitative. The Americans are using quantitative. You just put in quantitative. Yet, you're talking about qualitative. I see one in a million for average number of critical group. That's very quantitative. And the increase in cancer, that's very quantitative.

So what is the qualitative stuff you keep talking about and how do you derive from those numbers down to the technical where they are a release limit? I mean I don't understand this table. I don't understand the percentage. I don't understand anything about this table.

DR. RZENTKOWSKI: Quantitative means how we demonstrated the general health objectives are being met, and quantitative, it means that the incremental contribution to the public risk from nuclear accident is less than 1 percent.

So let's say that the risk of -- the public is subject to risk in that approximately one in 1,000 members of the public dies as a result of cancer. So now, an incremental risk is 1 percent of that risk. So that means one in 100,000 will die as a result of risk posed by the operation of the facility.

However, this risk is only calculated for the critical members of the public. So that means people living in the vicinity of the plant, specifically the person at the site boundary. So that's how this is calculated.

THE PRESIDENT: We understand that. We understand the health goals but we don't understand how you go from the health goals to 10 to the minus 4, 10 to the minus 5.

DR. RZENTKOWSKI: Because we just explained that health objectives is one in 100,000 and now the large release frequency safety goals is exactly the same, it's 10 to minus 5 for operating reactors.

So that means that the threshold, or the frequency threshold, and the release limits established for the large release protect the public from early and delayed fatalities because the evacuation will take place.

And also, the releases are limited to the value, which will only trigger either a temporary or a short-term evacuation but not the permanent one to reduce societal impacts to the extent possible.

So you can see those values are exactly the same in terms of numerics.

THE PRESIDENT: Industry?

DR. VECCHIARELLI: For the record, Jack Vecchiarelli.

Along the same lines as Dr. Rzentkowski was explaining, I'll put it in my own words, the quantitative health objectives are based on a certain percentage of the background risk. Let's take delayed fatalities, cancer fatalities. We're aiming for doing better than 1 percent of the background. The background risk is one in a 1,000 people dying per year due to cancer. We have a quantitative health objective that says a nuclear accident should contribute that 1 percent of that, so that's one in 100,000.

What we've done with the surrogate safety goals such as the large release frequency is we have set a more stringent goal on the actual large release which does not necessarily result in a cancer fatality. We've set that at the same frequency limit as the health risk.

So by focusing on avoiding long-term relocation or prolonged evacuation of the public, as that being the threshold at the same frequency as the health objective, meeting the surrogate will meet the health objective.

THE PRESIDENT: Ms Velshi.

MEMBER VELSHI: I think I kind of follow that but I'll switch gears.

Tell me why Level 3 PSAs are not done. I understand there's a -- you mentioned there's a lot of uncertainty but isn't that what really then tells you so what's the impact of these releases?

And I see in the table provided in one of the slides where the timeline of the different PSAs, it looks like a Level 3 PSA has never been attempted here and if you can correct me on that too.

DR. VECCHIARELLI: For the record, Jack Vecchiarelli.

The problem with the Level 3 PSAs is that you have to model a number of factors that have a great deal of uncertainty associated with them, such as the weather conditions, the movement of people through protective measures, and you're modelling this for many, many years, and what happens to radionuclides in the food chain. You're modelling the different pathways throughout the food chain, how people live and consume the food where land has been contaminated. This is ultimately what you would have to model to assess what the direct health risk is.

By focusing on the surrogates, what we've done is said, all right, we'll be more stringent, we're going to focus on -- we don't want an extensive societal disruption, so you've really lowered the bar, and you set the same frequency limit on that.

And the degree of uncertainty in getting through the Level 1 and Level 2 PSAs to be able to show that those safety goals are met do not have as much uncertainty as having to go through a full Level 3 PSA.

MEMBER VELSHI: Has anyone ever attempted a Level 3 PSA?

MR. MULLIN: Derek Mullin, for the record, from New Brunswick Power.

As a part of plant refurbishment, we went one step further beyond the Level 2 PSA and did some offsite consequence analysis.

Essentially, it's a very limited Level 3 PSA. We did not do a full-blown.

We looked at three -- basically three dominant sequences that would either lead to a containment bypass event, an event that would bypass containment release radiation. We looked at early containment failures if there was a failure of, say, a containment isolation valve and we were having a direct release. And we also looked at severe accidents that led to a consequential containment failure.

And we looked at those types of things. Again, there's a lot of assumptions that need to be made in terms of atmosphere conditions, you're looking at food and ingestion pathways, and you're looking at ultimately what the early and late fatalities would look like from a frequency perspective.

We did those calculations and I can't remember exactly for early or late where it landed but it was in the order of two to four orders of magnitude below these types of health objectives.

So we did go down that path and the reason we went down that path was not so much for the comparison against the safety goals but it was to use that kind of information in the context of future benefit-cost assessments and valuations and for assisting us with emergency planning purposes.

Thank you.

MEMBER VELSHI: Thank you.

And do you know if anyone outside Canada has done Level 3s or do they do that routinely?

MR. FRAPPIER: Gerry Frappier, Director General of Assessment and Analysis.

The U.S. NRC has a major project on the go right now where they are going to take a look at Level 3 PSAs and have a -- it's a research project, if you like, to try to come to the conclusion as to whether we've moved along enough for them to be practical to be used again. I'm not sure on the dates on that but maybe Smain or Yolande might now.

MS AKL: Yolande Akl, Director of Probabilistic Safety Assessment and Reliability Division.

The project that the U.S. NRC has will be completed by 2016 and given that there is no standard internationally for PSA Level 3 they will be writing their own standards to try for one application for one of their sites, the focal site. So that's where they are right now.

THE PRESIDENT: Can I ask a silly question again? If nobody is doing it, why does it exist? Who is demanding PSA Level 3s? Does the IAEA think it's a good thing to do eventually? Who is the driver for having a PSA? If nobody is doing it, where does it come from?

MS AKL: The IAEA now is starting a new project to look -- there was an old standard for PSA Level 3 many years ago, but for improvement now, we need to look at it again and improve that standard. So the IAEA has a working group on Level 3 PSA to provide a new document and new guideline.

THE PRESIDENT: But right now it's not a guideline that they recommend regulators follow?

MS AKL: That's correct.

THE PRESIDENT: Thank you.

Ms Velshi.

MEMBER VELSHI: It wasn't clear to me around aggregation of different PSAs. Is that done or isn't that done, Dr. Rzentkowski? I thought you said maybe December 2014. If you can confirm.

DR. RZENTKOWSKI: By December 2014 it's our objective to establish our regulatory position on integration. So this may be applicable to both the methodology and also what is acceptable to us. Because obviously using the same safety goals on a per unit basis and per station basis may not be appropriate.

MEMBER VELSHI: So currently the stations have for whatever internal/external events their PSA results but those don't get aggregated in any way, so it's hard to compare one against your target or the limit.

Presumably, aggregation doesn't just mean summing them up. So how do you know how close you are to your target or your limit if you don't know what the overall risk is?

DR. RZENTKOWSKI: Of course, we do. There's direct aggregation, to be honest, for our own knowledge.

But we use this as a safety indicator, not as the risk meter, which is guiding our decisions, because, as a safety indicator, it's a very useful input into our risk-informed decision-making process and that's how we utilize currently aggregation of individual risks because we do realize that there's a certain bias in aggregation, especially uncertainties related to calculation of external event sequences are tremendous.

You can imagine that we are trying to extrapolate this analysis going back, into, maybe 10,000 years -- what happened once in 10,000 years -- but we have historical information only for maybe two or 300 years. So that's the reason why uncertainties are tremendous and we have to have a better understanding how they can be treated or projected onto the internal events, so that from a mathematical standpoint it makes sense.

MS SWAMI: Laurie Swami, for the record. Perhaps I could just add to that that we know our plants are safe because we have multiple ways of assessing the safety of our plant.

The PSA is one approach and we compare that to the safety goal limits and targets on a regular basis. So, as I say, we know that it's safe because of the multiple ways we look at it, as we have described in our presentation.

MEMBER VELSHI: And do your PSA results vary from reactor to reactor on a given site?

DR. VECCHIARELLI: For the record, Jack Vecchiarelli. Risk is reactor specific, so it does depend on the station design and the geographical location what potential external hazards is the plant exposed to. So it does vary.

MEMBER VELSHI: No, I meant on a given site from one reactor to another.

MR. SANTINI: Miguel Santini, for the record. Basically the PSA is done for a certain design so you have a PSA for Pickering B, Units 5 to 8, one PSA for Units 1 and 4 at the same site.

MEMBER VELSHI: So you don't necessarily go into as much detail as to what is the maintenance history of all these components at this particular unit compared to another one on that same site then?

DR. VECCHIARELLI: For the record, Jack Vecchiarelli. All of the data collected from all the units are considered. We choose one reference unit to conduct the per-unit safety goal assessment.

THE PRESIDENT: Thank you. Monsieur Harvey...?

MEMBRE HARVEY : Merci, Monsieur le Président. We have been talking of PSA 1 and 2 and 3 for a certain period of time and I felt that the Level 3 was missing in fact, but realized today that it doesn't exist and it's not something missing, it's something which is not there.

So for the public it's quite different to say you don't achieve No. 3, but if it doesn't exist you don't need to.

In your document -- I'm talking to staff -- you mentioned that you don't need a Level 3 to guarantee that the plant is secure and that the population won't be hurt, so what are we looking for? What will be the final picture of Level 3; what kind of data, what kind of additional security?

DR. RZENTKOWSKI: Currently, what we have done, for the large release frequency safety goal, we established the release limit and this is unique to the Canadian framework. This is precisely to establish the condition which will trigger the evacuation of the population -- the members of the population, and this way we believe that the population is protected because we established a very clear trigger in terms of the frequency and the release limit.

In terms of the Level 3, it allows us to predict the health effects on the public and from that standpoint it could be very useful information, but it's not necessary for the protection of the public. That's how we look at this right now.

MR. FRAPPIER: If I could perhaps add a bit to that? Gerry Frappier, Director General Assessment and Analysis.

I think it is very important in this conversation for us to realize it's not just the PSA that we use to determine safety and any potential effects on the population. So we haven't talked about the safety report, which is the main tool that is used in reviewing the design of a reactor using deterministic approaches to ensure that there is no off-site releases that are going to occur, that are going to give it doses of any magnitude that would be of a health concern.

And so the PSA is an evolving tool, it has a really good track record of identifying vulnerabilities and allowing cost-benefit analysis to see how much you reduce your risk if you do certain improvements, but it is not the only tool that is used and it hasn't -- that's why the Level 3 has not really been necessary.

MEMBER HARVEY: In the Level 3 PSA, will the multi-unit site be taken into account in that Level 3 or something else?

DR. RZENTKOWSKI: The overall objective of Level 3 is to predict the health effects on the public. So, of course, the entire site has to be considered because looking on the per-unit basis may not be the right reflection of the reality under accident condition.

MEMBER HARVEY: So such a model would be very sophisticated and complicated because you need to have Level 1 for Level 2 and if you have four units you need to have four Level 1s, four Level 2s and this is quite a lot of data and complexity.

DR. RZENTKOWSKI: Absolutely. And calculation itself would be extremely complex and subject to uncertainty. So that means there will be a lot of subjectivity in interpretation of the data received. That's the reason why we believe that the approach taken here to establish the release limit for the large release safety goal is better because it protects the society at large and at the same time it is not subject to uncertainty, it is very clear. The release limit cannot exceed one percent of the reactor -- of the Cesium contained in the reactor -- Cesium-137 contained in the reactor core, and the probability cannot exceed 1:1 million years for new power plants and 1:100,000 years for operating facilities.

THE PRESIDENT: I love that characterization, I understand the characterization. What I don't understand then, how are you going to prove to me that the whole site is designed to meet those criteria? That is what we are talking about, if I understand correctly.

I like the way you characterize from the public perspective, that is, the health limit or the goal that we are assuming if we live beside a nuclear power plant that's the risk we take, we are willing to take.

So then the question is, on a

particular site how do you prove to us -- and I don't care if it's a combination of deterministic, probabilistic -- how do you prove to us that you actually meet those health limits?

DR. RZENTKOWSKI: That is the objective of the work we proposed here today, but it has to be understood that for the external hazards, or external events the frequency doesn't change, the consequence may change. So the risk will be slightly higher, but nevertheless, the frequency would be exactly the same as the one for the single unit.

For internal events it's a different story, but the consequences of the internal events are less severe because of the core damage frequency which will limit those kind of event sequences by an order of magnitude.

So I know this is difficult to explain from the public standpoint, but I really believe that the approach taken by the CNSC is extremely pragmatic, setting the frequency for the large release and the limit for the large release at the same time.

MR. SAUNDERS: Dangerous territory, but I am going to give it a try. Frank

Saunders, for the record.

So going back to Mr. Frappier's reference to the safety report, the safety report is deterministic in nature and it provides the accident sequences which the plants are designed to survive and to work against.

There is a health study that is associated with the safety report that actually looks at each and every accident sequence, the isotopes that would be released, the impact that would have on what we call the critical group, which is a group of people living right at the fence line essentially and the most -- the area with the highest concentration of that isotope and that is a very conservative deterministic way of doing it.

So each one of those sequences then must meet a release -- and this is where the notion of large release comes from -- must meet a release requirement that wouldn't expose that member of the public to those kind of health effects. So it's a very deterministic method that it is. And we say we don't do Level 3 PSAs, but we do health studies and each siting of a reactor has a health study and a safety report that's associated with it and you can go back and read those and we update them from time to time.

So there is something that is kind of like a Level 3, but it's done on a very deterministic basis and so it assumes -- it just takes the worst-case scenario and says what would that look like, prevent the worst case and you will prevent the rest. So it is a simpler version, but the tie is very direct. We just don't write it down that way because, as you can see, it is hard for people to understand how we got there, so we back up to large release frequency which we can measure easier and demonstrate easier, but that frequency really comes from that exposure. That's where it comes from.

THE PRESIDENT: Okay. Monsieur Harvey...? Mr. Tolgyesi...?

MEMBRE TOLGYESI : Merci, Monsieur le Président. On slide 24 you are talking about how different levels of release and uncertainty in PSAs. What do you consider as level of release, but level of uncertainty?

MR. SAUNDERS: This is really primarily looking at the difference between internal equipment-based modelling and external event modelling. So to calculate those probability we actually model the systems in the plants. So we model the pumps, we model the pipes, we model all those things and the computer runs, you know, thousands, millions of runs to estimate what that failure frequency might look like.

But we know that design very exactly, we know its performance very well, we know what the failure rates on that kind of equipment are, both from our own experience and from others, so you can actually model that and have a very accurate answer about, on average at least, what the failure rate would look like.

When you get into external events and weather it's much more difficult. First off, it is much harder to predict the event itself, you don't know where it will come from, what direction the wind might blow, how long it will blow, you know, will it be more focused if was a tornado, will it hit Unit 1 or Unit 6, right, so these things are a lot harder to predict accurately.

And even once you predict the weather event, just how it will impact the station is then also somewhat unpredictable because you don't know exactly what it will do on a particular thing. So although you can estimate it, it has not got the accuracy that an actual equipment-based PSA would have.

So it is predictive, it gives you an indication into the future, but, you know, if you look at the error bars on it they would be bigger than they will be on equipment. That's really the sort of extremes there. I mean you can see a little bit of that in seismic, right, I mean you will have an earthquake in one area, Level 6 and nothing will happen, and somewhere else things will fall down. So it will all vary then on the geology and other things, so it is a little harder to be accurate in external events than it is on the internal stuff.

MEMBER TOLGYESI: So the probability will be -- how do you call that? In French you say aléatoire. Aleatory, you could say that?

MR. SAUNDERS: I think so.

MEMBER TOLGYESI: Yes. Mrs. Swami, you said that PSA is one of several ways to evaluate the level of safe operations, okay. So when you compare that, those several other ones to PSA, what is the difference? Is there a discrepancy?

MS SWAMI: Laurie Swami, for the record. I didn't mean to imply you compared one to the other so you would be looking for a discrepancy. Mr. Saunders described how we have deterministic safety analysis in our safety report, that is one method that we would use. We look at how the plant is operating, we look at the PSA as one contributor to that.

So what I was trying to express was, there are multiple things that we do to ensure that our plants are safe to operate. This is one of the tools that we used to do that.

I think Dr. Vecchiarelli also described in his discussion today also how we use that when we're looking at different operating modes, when we are considering equipment that might be out of service. So that is -- a PSA has that application as well. So it is one of multiple tools that are used to help us and ensure that we are operating safely.

MEMBER TOLGYESI: So for the public in general, you know, who is looking and listening, I say how safe is your operation, and you are saying that our PSA gives us that it is quite safe for this and this reasons.

And when they ask, what about the other ones, you know, how far you are sure of this PSA? So that's why I'm talking about the discrepancy. How do you explain that, totally we are doing really well and there are several things which are proving that, but there are several things which you calculate or evaluate separate, different ways, okay.

So what is a common result after, because you should come to something which you will say, okay, we are safe.

MS SWAMI: Laurie Swami, for the record. So perhaps I will step back and talk a little bit about the safety and control areas.

When we look at the plant operation we look at multiple factors. So one of those would be our management system, so we have in place procedures and programs so that we operate within the balance of the safe operating envelope. So that would be one of the safety and control areas that we look at.

So if you have 14 safety and

control areas, within each one of those there are various things that we manage to ensure safe operation. So I mentioned management system, I would look also to our training and qualification program for our licenced staff. We spend a great deal of effort making sure that our trained and qualified staff are available to also operate our plants per procedure to be able to use their knowledge and experience and training to operate the plant effectively.

So all of those things that we hear about regularly in front of the Commission, each one of those elements contributes to safe operation.

Within the design feature we have the design itself, the requirements of the design, we have the operating limits that have been established, we have the operating policies and procedures that we operate within. So many factors are used in operation and licensing of our facilities.

So all of those are encompassed in how we determine if it's safe to operate and we consider all of those things on a regular basis.

MEMBER TOLGYESI: You know, we

were talking about PSA Level 3. What is the other industry or sector -- I mean chemical, because they do also have these materials. Do they have these PSAs Level 1 and 2 and 3s also or nuclear industry is the only one who is using this?

--- Pause

DR. VECCHIARELLI: For the record, Jack Vecchiarelli. In general there are probabilistic-type risk assessments that are performed outside of the nuclear industry, petrochemical industries, et cetera. There are places like that that use it, including aeronautical industry.

MEMBER TOLGYESI: Do you have to add something, staff? No?

DR. RZENTKOWSKI: I would like to go to the question, because this was very important -- to your previous question, it was extremely important.

How do we know that the public is protected? How we know, because for the events or incidents that may happen during normal operation of the plants the release limits are limited to .5 mSv, so the public is protected.

For the design basis accident

which may happen with the frequency between 1:100 years to 1:100,000 years, the release limits are limited to 20 mSv, so the public is protected.

Safety goals extend the design basis envelope to include severe accidents, specifically to make sure that their likelihood is limited to as low as practical and also to mitigate the consequences should an accident happen. So that is the objective of the safety goal, is to extend that design basis envelope below 1:100,000 years. Okay.

THE PRESIDENT: Okay. Anything else? Oh, Dr. McEwan, right. Sorry.

MEMBER McEWAN: That's fine. I know my place. I'm not sure this is a valid question, but I'm trying to understand the output of the safety analysis.

So if you take a PSA, as we are currently doing it, and you apply it to what happened in Chernobyl or Fukushima or Three Mile Island, how would the output of the PSA, as we currently conduct it, have reduced the likelihood of one of those events occurring?

MR. SAUNDERS: Yes, I think we can look at a couple of different ways of answering that question. But if you go to TMI as a starting place and you look at the operator error in essence that caused that, what a proper PSA would show you there is, it would indicate that there is a potential weakness in the design at that point that perhaps should be dealt with, right. And what the PSA does by modelling the systems very carefully, it shows you where you have things like single point of failures or critical components that are more likely to fail or operating errors that are more likely to create significant outcomes.

So the PSA model, if it is modelled properly, will show you where those kind of weaknesses exist. In effect, that is really the major benefit out of these models, is it allows us to look at different ways things can fail, understand what the major contributors to those failures are, whether it is a single piece of equipment, whether it is people or whether it is a loss of water, and then we can say, well, if half of the contribution is in this area what can we do to reduce that.

So that is really what it PSA does for you.

Chernobyl is the same thing, very simple in Chernobyl. They took the reactor outside its design basis. The PSA would have showed you immediately that that is a very wrong thing to do and that, you know, squishing all your flux down to the bottom part of the core was going to cause you some real problems. It would have predicted that event without question. I don't think they really needed a PSA, any scientist would have predicted that event.

And again, the same sort of thing really on Fukushima. There was a probability there which is understood, a very small probability, but, as we said, you know, risk is about probability times consequence, and so what is the consequence, a very low probability in that case, but a fairly significant consequence. You could have predicted some of the things like putting switchgear below ground is not a good thing to do if there is some risk of flood and maybe you want to have an alternate way.

The only answer there wasn't to build a higher wall -- I'm not really a structural engineer, so I don't know if you can realistically build a 50-foot wall that will hold back the sea in that circumstance, but certainly you could say, how what I make sure I had cooling to the reactor, how would I make sure I can add power.

And that is essentially what we have done with the emergency mitigating equipment, is found a portable way of doing that which we can keep out of harm's way and then we can deploy if we need to, even though the probabilities of those things in Canada are way, way less than they would be in many parts of the world, right.

So yes, PSA shows you those kinds of sequences and that is in reality where the value is. The number is just a quantification at the end which helps you assess the overall design, but from an operator point of view it is your knowledge of the individual sequences that are actually the most important piece.

THE PRESIDENT: Anybody else? Okay. Let me just make a couple of points. I think that -- I understand how complicated this subject is, but the issue in front of us is your own creation, the industry's own creation and the regulator's own creation. I can't believe that all those years site PSA was not something that, you know, we start looking at it a lot, lot earlier than right now and you didn't have to wait for Fukushima to all of a sudden everybody put in the whole site PSA and safety to the fore.

But then the problem, as I see it, is we have done everything on a per unit, per year PSA and then we couldn't figure out how to do it -- how to take our per unit into a multiple unit, actually do the math. And that's -- for somebody on the outside listening and he said, okay, the per unit is one and you have eight, so it's eight times.

Now, I know you guys are discounting this, but then you didn't help yourself when you weren't able to integrate the post-Fukushima mitigation, the emergency management into the analysis.

So there are a lot of things here that doesn't give -- give the impression that we don't know how to calculate it and, therefore, we cannot give the assurance that everything is okay.

I think that issue is still in front of us and, I mean, I was fascinated to see that you actually did a first approximation and in your slide 21 somebody calculated the risk reduction for EMEs at 2:10. So somebody must have done a back-of-the-envelope -- you don't have to run the whole model, you can run the worst-case scenario and come up with some binding parameters that may give you some comfort level that the site as a whole is safe.

So our dilemma is when we get to a site-by-site hearing, and we are going to have three of them coming up, we are going to keep asking the question: show us that it is safe for the whole site.

So I'm not asking a question, I am making a statement. But any comments, final, to close this particular file, please feel free. I see somebody is marching from the back.

MR. SANTINI: Miguel Santini, for the record.

THE PRESIDENT: Wait, wait. I think I recognize --

MR. ELLIOTT: Yes, good afternoon. Mark Elliott, Chief Nuclear Engineer for Ontario Power Generation.

We understand the challenge that you have posed, that the Commissioners have posed starting last summer, we accept the challenge. We are going to do the work to produce the whole site risk assessment, the whole site safety goals. The dates on the slide are a commitment, so we are rolling up our sleeves and we are going to do it.

THE PRESIDENT: Thank you. Staff, you were going to say something?

MR. SANTINI: Miguel Santini, for the record. I just wanted to comment that we have already submitted the staff CMD for the whole point for Pickering and we have put those numbers for the EME improvements on the results of the safety goals for Pickering and those are numbers calculated through PSA, those are not back-of-the envelope numbers.

THE PRESIDENT: There you go. I'm very happy to hear that you can do it faster than 2017 numbers. I'm looking forward to reading it. Anybody else?

DR. RZENTKOWSKI: Yes, Greg Rzentkowski, for the record. I would like also to mention that there is a logic behind that what we have done up to this point in time.

We developed the safety goals as a design requirement for the power plants and that's why the safety goals are defined on a per unit basis, because that is the only way how they can be applied. Now, the siting issues should be considered in the siting document, not in the design requirements documents for power plants because if we use siting safety goals for the design of power plants, we will arrive at a different design depending on the number of plants on a given site.

So we have to be very careful how we approach it and how this will be embedded in the regulatory framework.

THE PRESIDENT: Thank you. Anything else? Okay, thank you. Thank you very much.

--- Pause

THE PRESIDENT: I will take a chance here while Dr. McEwan is not here.

6.1 Fukushima Omnibus REGDOC Amendments Project / Projet omnibus de modifications des documents d'application REGDOCs relatives à Fukushima

THE PRESIDENT: Okay. The next item on the Agenda is on the Fukushima Omnibus REGDOC Amendments Project as outlined in CMD 14-M17 and 14-M17.A, and I understand that Mr.

Torrie will make the presentation. Please proceed.

CMD 14-M17 / CMD 14-M17.A

Oral presentation by CNSC Staff

MR. TORRIE: Good morning, Mr. President, Members of the Commission. My name is Brian Torrie, I am the Director General of the Regulatory Policy Directorate. With me today are Mr. Gerry Frappier, Director General, the Directorate of Assessment and Analysis, Dr. Greg Rzentkowski, Director General of the Directorate of Power Reactor Regulations, and Mr. Colin Moses, Director of the Regulatory Framework Division. As well, we are joined by an extensive team of CNSC staff who are involved in this project.

We are here today to present the final two documents in the Fukushima Omnibus Amendments Project, a project that involved targeted amendments to a number of existing documents in our regulatory framework to address lessons learned from the Fukushima accident.

As was reported to you at the meeting in August, 2013, this project is a key element of our overall regulatory framework improvements as part of the Fukushima Staff Action Plan. Today we will recap the general background on the project and the unique approach we have taken for public consultation, outline the general feedback we received from our stakeholders focusing on the results of an additional round of consultations and go on to discuss those specific improvements that are proposed for each of the two remaining regulatory documents you have before you today.

Finally, we will conclude the presentation with a discussion on implementation of these improvements, outlining CNSC staff's conclusions and recommendations.

The purpose of our presentation this afternoon is to request your approval to publish the following regulatory documents: REGDOC-2.4.1, Deterministic Safety Analysis; REGDOC-2.4.2, Probabilistic Safety Assessment for Nuclear Power Plants.

These documents include proposed amendments to existing regulatory documents, standards and guides to address the lessons learned from the event at Fukushima.

You will recall that following the presentation of the Commission in August, 2013, stakeholders were provided an opportunity to comment on the drafts through a second round of consultations. These comments will be addressed today. In general, the changes made to the documents as a result of this additional consultation have helped to clarify the intent of our regulatory expectations, but have not resulted in substantive changes to the documents' scope and approach. Following completion of this consultation, we received some additional feedback from stakeholders which will also be addressed.

At this point I will turn to Mr. Colin Moses, Director of the Regulatory Framework Division, to provide you with further details on consultations and development of the documents in this project.

MR. MOSES: Thank you. As was mentioned by Mr. Torrie, these documents were previously discussed with the Commission in August, 2013 as outlined in CMD 13-M35. At the time, the Commission approved publication of two documents in the operating performance and environmental protection series shown here.

However, further to a request from stakeholders, the Commission asked CNSC staff to conduct an additional round of consultation on REGDOCs 2.4.1 and 2.4.2, both in the safety analysis series of the CNSC's regulatory document framework. Today, after providing a general overview of the scope of the proposed amendments, we will focus on the results of this additional consultation.

The Fukushima Omnibus Project was focused on bringing targeted amendments to existing publications which have been consolidated into two documents consistent with the regulatory document framework shown on the previous slide. For ease of reference, this table maps current documents to the proposed publications.

REGDOC-2.4.1, Deterministic Safety Analysis, combines three existing documents into one. It consists of Part 1 that addresses nuclear power plants and Part 2 for small reactor facilities.

The final draft document also includes content from GD-310, Guidance on Safety Analysis for Nuclear Power Plants, which was originally published in March, 2012.

REGDOC-2.4.2, Probabilistic Safety Assessment for Nuclear Power Plants, includes amendments to existing requirements that are specified in regulatory document S-294. As a result of additional feedback from stakeholders, new guidance has also been included, as appropriate, which helps to clarify the intent of certain requirements.

Before going any further, I will provide you with an overview of the Omnibus Project. Following the extensive review of the CNSC's Fukushima Task Force, the Task Force concluded that a framework was robust and comprehensive, however, improvements were identified to further enhance our regulatory expectations. The issues identified by the Task Force were in many cases specific and cross-cutting, impacting content from a series of existing documents. This table illustrates some key lessons learned from the Task Force showing where they were addressed in a proposed document. As a result of these cross-cutting issues, CNSC staff took the approach of managing these improvements as part of a single omnibus amendment project. This was done to ensure a consistent approach to addressing these lessons across our entire regulatory framework.

For reference, we have also added a column which shows how REGDOC-2.5.2 for Design of Nuclear Power Plants also aligns with the identifying Fukushima issues. This project will be discussed further during the next agenda item today.

The CNSC has in place a practice of regularly reviewing and updating all of the elements of our regulatory framework, which allows us to ensure our requirements continue to reflect best practices in nuclear safety and the latest developments in nuclear regulation. The Fukushima Omnibus Project is above and beyond this regular process, bringing targeted improvements to our regulatory framework that are directly related to Fukushima lessons learned, as well as some specific improvements to our ensure documents remain current.

Despite the proposed changes being focused only on selected and targeted improvements, CNSC staff still followed our standard public consultation process for the earlier part of this project. Proposed changes were released for public consultation for an over 60-day consultation period from July to September of 2012. When released, the notice was posted on the CNSC's website and a notification was sent to all subscribers to our e-mail notifications, which now includes over 2,300 subscribers.

After the August, 2013 Commission meeting, an additional round of consultation was granted to interested stakeholders with the intent of giving them more time to review the proposed draft documents and, where necessary, to seek additional clarification of the CNSC's regulatory expectations. The second round of consultation was granted to interested stakeholders with the intent of giving them more time to review the proposed draft documents and, where necessary, to seek additional clarification of the CNSC's regulatory expectations.

The second round consultations took place from August to September of 2013 and the results of the comment dispositions were sent by e-mail to interested parties on January 28, 2014, along with updated drafts and the second round comment tables. The e-mail package permitted stakeholders to review the materials and several took advantage of the time to provide further observations or questions. Staff have reviewed these as well and have included an additional table with your material, updating the documents as appropriate.

The second round of consultation was limited to those who had participated in the initial consultation and we received 47 comments in all from six reviewers. It should be noted that one additional stakeholder, Greenpeace, also provided comments. These comments were taken into account as we finalized the draft documents.

I will be discussing the specific comments in the subsequent slides, however, it is worthy to note that some new issues were raised that had not previously been identified, primarily because they were regarding sections of the publications that were not part of the proposed revisions. Regardless, CNSC staff are in agreement that these comments, for the most part, helped to clarify our intent and have accepted or adapted most suggestions as appropriate.

Many of the comments received focused on questioning the consistency across different regulatory documents relating primarily to definitions and terminology, as well as some of the requirements being proposed through proposed revisions to the design document. CNSC staff fully agree with the need to ensure a consistent approach across all documents and, as a result of this feedback, have assembled a cross-functional team to address the identified issues.

As a result of this review, a number of changes were made to address historical divergence between the definitions in different documents. The review also identified some common issues that revolved around application of technical issues between new and existing plants that would benefit from clarifications. For example, to ensure consistent understanding of terms used in REGDOC-2.5.2 for design, a figure depicting the relationship between design basis accidents, beyond design basis accidents and design extension conditions was included in the safety analysis document.

In addition, commenters suggested deleting references in REGDOC-2.4.1 for safety analysis to design requirements, arguing that the operator action times in the safety analysis document are common practice and the reference to

REGDOC-2.5.2 for design could risk confusion. As a result, CNSC staff maintain the proposed reference, however, modified guidance in REGDOC-2.4.1 to provide further clarification about the effective date that is applicable for new nuclear power plants only.

In addition to providing feedback on the specific amendments being proposed by CNSC staff, some stakeholders offered suggestions for additional amendments or changes to the regulatory framework that they felt were warranted, specifically stakeholders reiterated that CNSC should finalize draft document RD-152. For background, staff had developed this document and released it for public comment, however, ultimately concluded that the document was not a necessary element of our regulatory framework for a number of reasons.

For example, the document was for general information purposes and did not propose to include any new requirements or guidance on licensee activities. In addition, many of the elements in the draft document are already found in existing documents or the licensing basis of existing facilities.

Subsequent to both rounds of comments, CNSC staff revisited this decision, but ultimately concluded that the rationale for termination of the project remained valid, noting in particular that a number of the elements that were included in RD-152 have since been incorporated into existing documents as appropriate. As an example, the documents for safety analysis and PSA now contain more information on objectives and provide additional regulatory guidance.

I will now speak to the specific comments received on the documents. To start, I will provide a summary of the key updates included in REGDOC-2.4.1 for Deterministic Safety Analysis. This document describes CNSC expectations for performing deterministic safety analysis, providing requirements and guidance for the selection of events, acceptance criteria against which to assess these events, and on acceptable analysis methods and documentation.

This document is divided into two parts, the first addressing regulatory expectations for nuclear power plants, and the second for small reactors. With this project, CNSC staff have proposed specific amendments to address lessons learned relating to accidents and events. Consideration of potential cliff edge effects, review of makeup water and reserve electricity capacities, review of low frequency events, as well as additional expectations for design basis accidents and beyond design basis accident analysis. These changes were equally applied to nuclear power plants and to small reactor expectations.

As you may recall, during the August Commission meeting, some stakeholders were concerned that text incorporated unchanged from the guidance document GD-310, which was originally published in March, 2012, could risk being interpreted as requirements in the future. While it is understood that guidance text in the document is indeed intended as guidance, in particular, through locating the text in specifically designated guidance sections, CNSC staff reviewed and confirmed that the intent of these statements was a guidance, adjusting the language as appropriate.

For example, words such as "shall", "has to" and "minimum expectations" have been deleted, clarified or replaced by language like "should" or "performance objectives".

Some stakeholders referring to recent developments in draft IAEA documents suggested changing the term "complementary design features" to "additional safety features". CNSC staff agrees that the terms may be used interchangeably and have added a note in the document to that effect. However, to minimize introducing additional inconsistency in our definitions, we maintain the use of the term "complementary design features" in the documents for safety analysis and PSA, as well as a nuclear power plant design as it was already being used in earlier documents.

Similar to feedback on REGDOC-2.5.2 for Design of Nuclear Reactor Facilities, commenters raised uncertainty as to how the safety analysis document will apply to new and existing nuclear power plants. To provide some clarity, CNSC staff updated the preface to these documents to align more consistently with REGDOC-2.5.2 Design of Nuclear Power Plants. Notes in the document for specific sections were also clarified to indicate that new nuclear power plants are those licensed after publication of REGDOC-2.5.2.

I think you are lost there. Sorry.

The second document being presented in REGDOC-2.4.2, Probabilistic Safety Assessment, this document defines regulatory expectations for the conduct of a probabilistic safety assessment for nuclear power plants. This project amends existing document S-294 and includes changes to address lessons learned from the Fukushima event, including to perform both Level 1 and Level 2 PSA with particular consideration of spent fuel pool and multi-unit events. It also clarifies external event analysis. Some additional changes were included to ensure our requirements continue to align with current international practice and to provide additional guidance to the users of this document.

As mentioned, commenters noted that guidance should be added to guide licensees on how to apply the document. Some even felt the previously issued document was unclear or lacked detail and guidance and imposed regulatory uncertainty on licensees. As a result, guidance has been added into REGDOC-2.4.2 where stakeholder comments suggested such clarification was needed. This provides guidance appropriate for the users of the document, that is, the practitioners who apply PSA.

For example, as a PSA document is objective-based and not intended to be prescriptive, licensees and practitioners are expected to identify and choose a PSA methodology that is appropriate and acceptable to their specific facility. Guidance to assess what the criteria for acceptable approach is should help in that regard. In addition, specific guidance was added to support the introduction of new material such as the objectives of PSA that are derived from international documents, fundamental safety objectives and site-specific methodologies. With the new guidance, explanatory text is being added to the preface to explain the approach, the application of the document, which also brought consistency across the safety analysis, PSA and design documents.

Some commenters reiterated their suggestion to require Level 3 PSA and the need to assess potential off-site consequences affecting the health and safety of Canadians in the environment, as well as to assess the cumulative radiological risk posed by the reactor site. Upon further consideration, and as you heard earlier today, CNSC staff have reconfirmed that no change is necessary to the requirements of this document. For regulatory purposes, the assessment, including the consequences evaluated as part of the Level 2 PSA, are sufficient and contain the necessary information to assess plant safety, providing insight into plant vulnerabilities and adequacy of design, operating procedures and mitigation provisions. Through the PSA, a limit is set on the frequency of radioactive release, which allows off-site consequences to be assessed deterministically by means of atmospheric dispersion models of radio nucleotides.

Regardless, CNSC staff are actively involved in the international effort to develop a consensus approach to Level 3 PSA and will keep abreast of developments in that regard.

Although commenters suggested adding requirements for cumulative radiological risk posed by multiple units at a site, no specific change for requirements were seen as necessary. One of the objectives of the Fukushima Omnibus Project is to include the need to consider multiple units at a site, so this aspect had already been added as a requirement.

However, in order to assist licensees on future developments, CNSC staff included guidance, noting that CNSC is currently reviewing methodologies for developing multi-unit PSA to evaluate the integrated risk. This guidance includes considerations in the development of risk metrics. As the CNSC continues to work with the international community, and the IAEA to develop risk metrics for consideration of the risk posed by multiple unit sites and to develop the associated multi-unit PSA methodology, these lessons would be integrated into the document.

Commenters suggested that the results of the probabilistic risk safety assessment should be publicly released and that public assessment risk should be added as a PSA objective. Public information is established in licensing as a requirement through another regulatory document, RDGD-99.3, Public Information and Disclosure Programs. However, CNSC staff have included guidance into REGDOC-2.4.2 for PSA that recommends public information include high-level summaries for PSA.

Subsequent suggestions from stakeholders for the potential inclusion of high-level methodologies and screening criteria were also added to the guidance. As noted, however, it is recognized that this release must be balanced with security, confidentiality and proprietary considerations.

Some commenters felt the requirement to seek CNSC acceptance and methodologies and computer codes for PSA should be removed. Other stakeholders felt the acceptance that -- felt the acceptance should be the responsibility of the Commission rather than CNSC staff.

The requirement for prior acceptance of the methodology and computer codes is current practice and is required through S-294. REGDOC-2.4.2 takes an objective-based approach rather than being prescriptive. This allows the licensee the flexibility to choose the most appropriate methodology based on the facility and its risk profile; however, this flexibility does require additional verification by the regulator.

In practice, experience has shown that CNSC staff involvement at this stage has been beneficial to assure a consistent level of methodologies and application. As the objectives of PSA and many methodologies are new, it is also important to guide licensees in their introduction and application. As a result, this requirement was retained with some additional guidance to emphasize a need to ensure the methodologies support PSA objectives.

Suggestions -- suggested amendments were also made to clarify expectations for management systems or quality assurance. However, stakeholders in their previous feedback noted that these changes were not directly related to Fukushima. Reviewing the proposed change and the intent behind it, staff were in agreement that the changes did not add substantive value to the document and were, in fact, redundant to existing management system requirements and the current licensing basis for nuclear power plants. There was no need to duplicate requirements in the PSA document that already existed elsewhere.

Because the existing reference

standards have since been superseded and in order to ensure the document remains accurate despite any developments in management system expectations, staff have removed all direct references to these standards and regulatory requirements, noting only that the applicable management system or quality assurance program expectations will be established on a licensing basis. Reference to the documents was maintained as guidance, however, primarily as a convenience for potential new license applicants who may not yet have an established licensing basis.

As noted, following completion of the additional consultation, the resultant drafts were circulated to stakeholders, some of whom chose to provide additional feedback. This feedback generally suggested minor changes to help clarify the intent for the users and most were accepted or adapted as appropriate. Stakeholders also suggested the development of a distinct CNSC glossary for regulatory documents. This suggestion has merit and CNSC staff will explore the feasibility of implementing the suggestion.

To summarize, CNSC staff have thoroughly reviewed, assessed and responded to all comments received during the multiple periods of public consultation. The full response to comments received during the additional consultation rounds are outlined in the detailed comment tables provided and a summary of the main themes coming out of all consultations has also been provided.

As appropriate, changes to the proposed amendments were made and integrated into the final consolidated draft documents you have before you today. In addition, further to the feedback received last August, CNSC staff have added a new step in our process to perform a second round of consultations if the feedback from stakeholders was substantive and resulted in significant changes. And as a minimum, to provide REGDOCS to all stakeholders well in adva-- to stakeholders well in advance of Commission meetings, giving them an opportunity to review CNSC staff's response to their input.

Should the Commission approve publication, the documents will be finalized and posted on the CNSC website, with a notice sent to all subscribers. Affected licenses and their associated License Conditions Handbooks would then be amended as applicable to reference the new documents, with specific implementation plans outlined in the LCHs as appropriate.

In conclusion, further to the thorough review of the CNSC Fukushima Task Force, the development of the CNSC Staff Action Plan and the work of CNSC staff to develop the regulatory expectations following a rigorous and extensive public consultation process, CNSC staff are of the opinion that the proposed revisions are appropriate and address all relevant lessons learned from the Fukushima nuclear accident, as identified by the CNSC Fukushima Task Force and the External Advisory Committee.

CNSC staff, therefore, recommends that the Commission approve the publication of the proposed regulatory documents.

This concludes CNSC staff's presentation on this matter and we remain available to address any questions you may have.

THE PRESIDENT: Thank you. Well, this document, those two documents have been consulted quite a bit over the last few years, so I would hope that most of the concerns have been addressed one way or another. Nevertheless, just if there are any lingering differences of opinion. We've got some industry people and some other interveners here, so I will recognize them and ask them whether they have any issues with what's being proposed and I'll start with a representative from industry. Anybody from industry who wants to share some comments?

MR. SAUNDERS: Good morning again. Afternoon now, I guess. Yeah, Frank Saunders for the record and my comments are actually Bruce Power specific in this case.

The issue here when we started this, the intention was to make a relatively quick change in these documents, which were reflective of the Fukushima changes we wanted to do. Along the way we decided to expand those changes somewhat. And it was our view when that happened that if we're going to expand the changes, then we ought to look at other problems with the document and resolve those as well.

And I think in particular, to emphasize where I'm coming from, if I could get you to turn to 2.4.2 section 4.7 is the best way for me to illustrate my concern. 4.7. It's page 3 in mine, but I'm not sure they're all the same.

So section -- so number 4.7.

THE PRESIDENT: Okay, we're on the same page.

MR. SAUNDERS: Okay. So section 4 of this document is the shell section, which means these are obligatory requirements that we must meet. And this document requires us to seek CNSC acceptance, in other words CNSC approval, of both the methodology and the computer codes that we were using.

In our view this is not sufficient clarity and direction of the requirements. It may have been sufficient in 2005 when we first issued this document, but we've been doing this a long time now. We should be able to write down the general requirements and the technical inputs that are necessary in such a way that a licensee can determine what is needed to be done, go and do that with a reasonable level of predictability on meeting the regulatory requirement. So without requiring work direction from CNSC staff, quite frankly.

So we don't agree that this document should simply say seek staff consensus. Either the document has enough information in it and this requirement is not there, CNSC staff reviews everything we submit at any rate, or the document should be amplified to make sure that the guidance is sufficient that we can do the work and be predictable about whether we're going to meet the requirement.

If you looked at the sister document to this 2.4.1 and looked at section 4.4, you'll find it's 20-some pages long for a deterministic-type approach. So quite a different document and I don't really want to propose that I want CNSC documents to be longer and more detailed because that's not really -- not really my intent, but I do think that this one lacks and then provides an arbitrary acceptance by CNSC staff which doesn't allow the predictability that licensees should have. And when you consider this area in particular, as we discussed earlier this morning, it's a lot of work, it costs a lot of money and it has a profound impact on the acceptance of the facility. So we think this needs to be spelled out better than it is here.

At the same time there are things in this document which are a definite improvement, which are helpful in the process, so we're somewhat loath to suggest that the document not be issued. What I would like, I guess, if I had my wish, would be that we would be able to get some kind of commitment over the next two years, that this -- that these requirements will be defined in such a way that we could understand them and predict them without having, as I say, a particular work direction from CNSC staff on how to do it.

You know, in the world, you know, we do this a lot to standards, technical standards, you know, because this is about the how; right? And technical standards are one way of doing it, you could do it in this document, but I really think they need to be defined and written down in a much clearer fashion.

THE PRESIDENT: Staff.

MR. SAUNDERS: And that's my comment.

THE PRESIDENT: Staff.

MR. FRAPPIER: Gerry Frappier, Director General of Assessment Analysis. I agree in general with the principle, if you like, that Mr. Saunders is talking about in the sense that it would be more appropriate to have a better definition, but as we talked about earlier this morning and even quite in a broader context, the methodologies themselves are changing quite quickly right now. There's a lot of innovation coming in. There's a lot of new aspects to it. And as was discussed, it's very costly to undertake these analysis, so we want to ensure that going forward we have an agreement as to how the analysis is going to be done before there is -- you know, before there is a whole bunch of expenditures that then make it more difficult to say we don't think you did the methodology just right.

We have been encouraging industry and we would certainly support the idea, as Mr. Saunders is mentioning, to have a technical standard that would talk a little bit more about exactly how to do these PSAs. If that were on the table, that would be good, but we don't have that right now. So we're in a bit of a flux with respect to probabilistic safety assessments and we think that this clause is important in the interim to ensure that there isn't a huge disconnect as to how these are being done.

As Mr. Saunders mentioned, in the

deterministic safety assessment world we have much, much more experience, there's a lot more clarity, I agree with him a lot more clarity as to what our requirements are. There's a lot more that can be learnt by just looking at the document. The PSA world, we're not quite there yet. I'm not sure if a two year time would be enough, but something of that nature should be done, but it cannot be done today and I think this clause is still necessary.

THE PRESIDENT: So this is a guidance clause, if I understand correctly. Did I get -- it is a guidance, isn't it? It's not a shall, you shall, you must have a pre-approval of CNSC on this?

MR. SAUNDERS: Yeah, no, section 4 is a shell section. It's a must do.

MR. FRAPPIER: It's Gerry Frappier again. I would point out that we do have some references that we point to that have been done that could be used as guidance written by other parties. And if you want more detail on that, we can -- that's ...

MS AKL: Yolande Akl for the record. There are two guidance documents from the

IAEA that are referenced in our document that give guidance on the methodology and how to perform a PSA.

THE PRESIDENT: Okay, so I misunderstood. I thought you have to do it, but I thought how you do it still has some flexibility built into it.

One other thing that I'd like to just get clarification, it is not difficult to amend, change, approve on an ongoing basis a document? Can somebody address that? Everybody believes that once those documents are out they are cast in concrete.

MR. MOSES: Colin Moses for the record. We've -- as we've discussed before with the Commission and made clear on our website and in engagement with our stakeholders, we welcome feedback on our documents at any time. If there's challenges in their implementation, we collect that feedback and we review it to see whether immediate changes are needed or whether it's an issue that we can sort of park and address the next time it comes up for regular revision.

MR. SAUNDERS: In general terms, I guess, I would object for any statement like that, any CNSC document that says you will only do what staff approves you to do. It amounts to work direction to the license, which in our view is not appropriate; right?

I do understand that it takes some time to do these things sometimes, to develop the requirements, but certainly in a number of areas here, Level 1 for sure, we know these requirements and we know how to write them down. Our fear is always, you know, that when things are written in that way, they're subject to misuse or mis-- they're rather arbitrary, people change, sections change, thoughts change and suddenly you can find yourself facing a whole different. And I -- we have, over my time in the nuclear industry, faced this problem. So we prefer that things be stated clearly, right, with clear objectives, clear requirements. And where there's more detail and guidance required, it should be provided either through a standard or something else and we should make sure it happens.

So as I said, we're not really trying to prevent this document from being issued, but I really do want to have some kind of a commitment that we want to let it stand in this form, that we will fix it and make sure the guidance is provided clearly so that, you know, anybody experienced in the field could pick up the document and understand what the requirements are.

THE PRESIDENT: Well, is there such a commitment? Are you willing to give such a commitment?

MR. FRAPPIER: Gerry Frappier for the record. We are certainly committed to doing that. And like I said, right now we're having discussions whether it's more appropriate to do a CSA standard that would say the how, which I think would be the more normal approach, if you like, and so that then becomes having the commitment from CSA and industry and all that to make it happen. Mr. Saunders has suggested the two years sort of time frame and that's probably appropriate from our perspective as well. If you want, we could report back to you in that time frame.

THE PRESIDENT: Okay. (off microphone) -- get back on that.

Any other issue from industry? Okay. I guess after two years, I would hope that we will get some sort of consensus.

We also are very happy to hear

that Greenpeace actually participated in this regulatory consultation and we have Mr. Stensil here to share his views on it.

MR. STENSIL: Yes, thank you, President Binder and commissioners.

I have two comments, so what I would suggest is we do them one by one with questions, if that's okay.

The first has to deal with transparency and disclosure of information. There's section 5 in the Guide that deals with this. I think the Commission needs to give some additional direction or clarification to staff and licensees regarding their obligations to release PSA information. This could either be done through a line in the Guide or in your minutes to this meeting.

And those two elements are -- with PSAs there's the issue of when they're released for public consultation or made publicly available. That's an important issue for non-industry stakeholders. And the second issue has to deal with how licensees have been using security concerns to block info related to public health risk that I think needs to be dealt with more clearly.

So if we're on the first issue, I have a very easy example to give you. The -- one of the reasons why I participated in this round is at the August meeting I was watching on the webcast and I heard the licensees trumpeting how they were making all this information publicly available. And I have to say the experience on the other side is quite different. These -- OPG, yes, has started to release PSA summaries. It's come with them being pushed, of course. But to give you an example, as you all know at the Pikering hearings last year, there was a great deal of controversy related to the PSA. That's why we just had this long discussion before the break. And one of the issues was that they hadn't provided the Pikering A PSA to the hearings. And the Commission, which I support, agreed with a request for ruling that NGOs put in asking for them to resubmit it this year.

We're now in the public consultation period for that meeting on May 7th and that document hasn't been released. So we have 30 days. These are complex documents. But if you're on intervener on the outside world, you need time to be able to comment on these things in a -- you know, in a way that provides good advice to the Commission, and right now it's been quite haphazard how the industry releases these documents. The other example would be the Darlington PSA, which we got at the beginning of a consultation period but wasn't at the beginning of the environmental assessment review. And as you know, there was a lot of questions about that document that could have affected the scope of the EA.

So my first point would be there needs to be something, whether it's in the document or whether it's in your -- the minutes of this meeting, on licensees who are moving to five-year licenses, for example, licensees should have their paperwork together before we go into a re-licensing period or an environmental assessment. That makes it better for the public to be able to actually look at these documents. I haven't seen that yet. So that's the first comment.

The second comment is a little more complex. Greenpeace requested wording in this REGDOC that would require information related to offsite health and environmental risks to be released. We also noted in the wording it should be released to provincial offsite emergency planning organizations. This was a reasonable request, but staff said no to putting that type of wording in.

Based on my experience, I think there needs to be more direction on this. I think there's only two of you left on the Commission that were around at the time, but in 2008 Greenpeace made a request for PSA information around the Pikering B re-licensing at that point, specifically related to source term information. So this is the information that's used to calculate offsite risks, test emergency planning. I sent correspondence to the Commission in the summer saying that ruling has been used in other fora to blank out all availability of information.

And to give you a couple of examples how OPG has used this to withhold source information that doesn't have security implications. One is an FOI request I asked for through a provincial FOI. It went to adjudication. FOI cited that decision to withhold the information. The former OPG engineer Frank

Greening, you may know from other instances, sent a note to the Information Commissioner that I provided the Commission saying how these arguments are all spurious and the information can be -- or equivalent information elsewhere can be found.

But what I've also gleaned through FOI and from staff at different agencies is that OPG has been refusing to release source term information on the large release accidents to both the CNSC staff and Emergency Management Ontario. And this is based on that assumption that goes back to 2008. For CNSC, as you may recall, staff undertook to do a large accident study and they're using something called a Generic Large Release Scenario, but they don't want to use this PSA information or -- allow CNSC staff to release it and they're using security concerns, and I think that needs to be rectified.

And the same thing, I understand, through Emergency Management Ontario. Good news, the province has decided to do a review of offsite plans, which has been requested to the Commission often, but the minister there has undertaken such a review. My understanding is OPG isn't releasing that source term to the province.

So I think it is in the public interest that this information be released. So I'd request on this one there be an additional line put in the document around information related to health and environmental risk for testing emergency plans and that that information be specifically given to provincial emergency management organizations. Thank you.

THE PRESIDENT: Okay. Let me start with staff. This is not a new issue about what is reasonably disclosable on PSA, so where are we on that and what is your view about the text you put in in guidance on public disclosure?

MR. MOSES: Sure. Colin Moses for the record.

As noted in the document and in our presentation, we set out extensive public information program requirements through a different regulatory document and that's consistent with the structure that we presented to the Commission before, to sort of group the common elements or related requirements into single documents.

The principles outlined in RD-99.3 require licensees to reach out to their stakeholder communities and identify the information that they want to hear about and the formats and structure for that information and to establish public information programs, and the CNSC oversees implementation of those programs.

But given the feedback we received from stakeholders and from the Commission in August 2013, we decided that it was appropriate in this case to include specific guidance suggesting the release of information related to PSA in this regulatory document. We included that guidance.

We received some additional feedback from our stakeholders during this additional round of consultation and made some minor adjustments specifically to highlight fault sequences and the summary of the results and the assumptions of the PSA.

MR. FRAPPIER: Gerry Frappier.

I agree with what Mr. Moses is saying, is that there is a document associated with release of public information. That's RD-99.3. This is a technical document and I think that we have added a section 5 here based on the comments that we received and we believe that from a technical document perspective that does provide sufficient guidance that says yes, this information should be released.

And the other aspects of when, how and things like that are more appropriate in the --

THE PRESIDENT: Well, we can test it.

Ms Swami, how will you interpret 99.3 and this section 5?

MS SWAMI: Laurie Swami for the record.

I won't try to interpret the guidance but I would clarify that OPG, through our work on PSAs, has been releasing summary reports. These are technical reports that are released, about 100-120 pages in length, that summarize the technical work that we've done on PSAs.

We recognize that there are security issues, that we cannot release the full PSAs, and so we've looked for a means of giving the information to the extent we can. That information has been made available when we complete our PSAs.

The requirement to complete the Pickering A PSA has just recently been done as required from the licence condition. We've done that. We are now in the preparation of the technical report. Of course, a technical report must be done in a rigorous manner, so we're going through that process and we'll publish it in April.

THE PRESIDENT: Okay. What about the second item on emergency, offsite, source term, all of this, where is that coming from?

MR. MOSES: Colin Moses for the record again.

Similar to the public information disclosure, we're working on a regulatory document right now to outline our expectations for emergency preparedness and response.

And so, information, for example, Provision of Information to Offsite Authorities, we just recently completed our public consultation on that document and that's one of the elements that we're looking at in the context to introduce our expectations around that provision of information.

So it would really -- if we're talking about emergency provisions, we'd be addressing it in that regulatory document as opposed to specific to the PSA regulatory document.

THE PRESIDENT: Mr. Stensil?

MR. STENSIL: Yeah, just to repeat, on the PSA information or the source term information, OPG has been saying consistently for the past four years that this information where you could third-party test the robustness of offsite emergency plans is too sensitive to be released to the public. They've been using a request for ruling from the Commission in a bad interpretation now for five years.

So if there doesn't need to be a statement in the Guide, as they mention off the top, in the minutes to this meeting if the Commission makes a statement on the expectations of issues security-related but things related to offsite planning and health and the environment that that should be released. That will help stakeholders going through Access to Information to be able to rebut that.

On the second issue, it's the same. While Ms Swami said, we're preparing the Pickering A PSA, stakeholders are in the problem that we'll have a week to look at it before writing written submissions and this has happened before. So again, if it's not in the Guide, just an expectation from the Commission will help future hearings and you'll get better advice.

THE PRESIDENT: I think you have a commitment here or a public announcement that it will be available sometime in April directly from the source. I thought it was better than saying it even in our minutes.

MR. STENSIL: The public has 30 days --

THE PRESIDENT: But it will be in our minutes as a fact.

MR. STENSIL: Okay. Yes. But the point is 30 days is not a lot of time to review the technical documents of this sort. So for future hearings, if OPG and Bruce Power and NB Power are aware that they should have it ready the day the CMDs come out, that would be very helpful.

THE PRESIDENT: You know, we also will have only 30 days to read this stuff. Anyhow, that's a comment.

Okay. Anything else on this?

MR. STENSIL: I had a second issue.

THE PRESIDENT: I thought the second issue was the offsite.

MR. STENSIL: No. It was a two-prong transparency issue.

THE PRESIDENT: Okay. Please quickly because we are running out of time.

MR. STENSIL: Okay. I'm sorry, this one may now be controversial after the debate you had in the previous information item.

My comment here is the current guide and review, while having been extensive, has not considered all the lessons from Fukushima regarding PSA. Specifically, the CNSC has yet to modernize safety goals, which was acknowledged in the last session, to address the risk posed by site-wide risk instead of individual reactors as well as the metrics of consequence that are used to judge social acceptability.

I would like the Commission to acknowledge this in the minutes of this meeting as well as -- so in a way, this is a temporary guide -- a commitment to consult the public on new site-wide goals, including metrics that will be used.

That is to say, we should

acknowledge that the current guide isn't a finalized response to Fukushima, it's a temporary guide.

To explain, as you know, issues related to the loopholes and weaknesses of PSA regarding Fukushima have been raised by Greenpeace during the Fukushima Action Plan Review, the Darlington hearings and the Pickering hearings last year. It was only at the Pickering hearings that we finally saw an acknowledgement of these issues with initial actions or undertakings to address them.

The Pickering Licence Condition Handbook states:

> "CNSC staff..."

And I thank the Commission for actually imposing those conditions.

> "CNSC staff are currently, jointly with industry, in consultation with the international community, developing concept-level metrics and/or redefining safety goals." (As read)

This was talked about in the last

session. So the idea of what is an acceptable risk to Canadian society is being developed right now but it's being done at the IAEA level and with OPG.

Safety goals are used to justify what you determine as reasonable risk to Canadian society under the *Nuclear Safety and Control Act*. Canadians should be involved in this.

First of all, and this came up a bit in the last section, I can tell staff have been very busy developing these goals through Access to Information and there were a few timelines that were stated, I think, in Mr. Rzentkowski's presentation and with OPG.

And here I have a presentation that CNSC did internally from September 2013 following the Pickering hearings and are timelines on both developing new safety goals and on the metrics, on which we haven't had a lot of discussion, and mid- to long-term action, develop safety goals by December 2014.

I don't see any period in there where the public is actually being asked: What do you view as an acceptable safety goal in the 21st century?

I've been trying to get other information about the correspondence with the IAEA. It's all being blocked.

THE PRESIDENT: Okay. Look, this is not the item we're dealing with right now. We're dealing with specific regulatory documents. Do you have a proposal?

On the development we're going to go through our normal process. Our normal process is that when we have a new regulatory amendment to the PSA, we do consult. So it doesn't have to be necessarily repeated. Consultation is part of the process.

Right now I'm interested in your view about those two documents please.

MR. STENSIL: Well, what I'm asking for here -- what I'm worried about is what happened with -- Monsieur Harvey, you'll remember this -- with life extensions of reactors in Canada. The process -- the oversight of that was basically industry-led. I'm worried about the development of these safety goals behind closed doors. When it comes out to public consultation, it will be a fait accompli without any input from Canadians on what those metrics are.

So the question of how much risk is acceptable, I would like in the minutes to this meeting an acknowledgement that this is a temporary guide, there will be revisions to it in the future, it is not the be-all, end-all of the Fukushima analysis, and I request that the Commission acknowledge there will be a future consultation period of safety goals as well as metrics.

The last discussion period, I didn't intervene at that point but a lot of that dates back from the 1950s when people are talking about health risk. I'll remind you that your Act that you work under that was changed in 2000 talks about human health and the environment, and in your safety goals at present there is no discussion of protection of the environment as a metric.

We need to have a more open debate about that. So I'd request in the minutes you at least acknowledge that there will be a consultation and that the documentation or consultations with staff will be made public. That's all.

THE PRESIDENT: Okay. Thank you.

Thank you for your intervention.

There is one more intervener that we would like to hear from and I understand he's coming to us online.

MR. ROUSE: Yes, but I'm on the phone.

THE PRESIDENT: Mr. Rouse?

MR. ROUSE: Hello. Can you hear me?

THE PRESIDENT: Yes, we can.

MR. ROUSE: Okay. Just a second.

Chris Rouse for the record.

Good afternoon, Commissioners.

I would like to thank you for this opportunity to speak to you regarding the CNSC staff's request for your approval of REGDOC-2.4.2, PSA for Nuclear Power Plants.

I have some very serious concerns with changes made in the last round of consultations --

THE PRESIDENT: Can you get closer to the mike, please?

MR. ROUSE: Oh, sorry.

I have some very serious concerns with changes made in the last round of consultations which are contrary to the lessons learned from Fukushima, and from the public's perspective it is very worrisome. It might not even be within the purview of the Commission to accept these changes.

My first concern is with section 3a "Objectives of the Probabilistic Safety Assessment." The original text was as follows:

> "to provide a systematic analysis to give confidence that the design will comply with the fundamental safety objectives; the fundamental safety objective, as established in IAEA N-SF-1, is to protect people and the environment from harmful effects of ionizing radiation."

After a suggestion by OPG, Bruce Power, AECL and Candu Energy, it was accepted by CNSC staff to change "comply" to "align." The change from comply to align is a significant weakening and would not be very enforceable. Comply is legally well defined.

After consultations with the Canadian Environmental Law Association they did a search and could not find any judicial interpretation of the term "align" or "alignment" in a statutory or regulatory context in Canada. Why are CNSC staff accepting wording suggestions from the licensees that have no legal meaning?

The IAEA document N-SF-1 is not a guidance document. It is the Fundamental Safety Principles on which the *Nuclear Safety and Control Act* is built upon to be consistent with Canada's international obligations when they signed onto the *United Nations Convention on Nuclear Safety*.

To protect people and the environment from harmful effects of ionizing radiation and the prevention of unreasonable risk to the environment and to the health and safety of persons are not guidance or something that should be followed, they are something that under the *Nuclear Safety and Control Act* shall be followed. The objectives of both the *Nuclear Safety and Control Act* and N-SF-1 are mandatory and not guidance.

I request that the Commission get legal advice if it is even within the purview of the Commission to make this change as I believe it is contrary to the requirements of the *Nuclear Safety and Control Act* and its international obligations for the *Convention of Nuclear Safety*.

My second concern is with section 4.7 Methodology and Computer Codes under the section Requirements for a PSA. The final version states:

> "Methodology and Computer Codes Seek CNSC acceptance of the methodology and computer codes to be used for the PSA before using them for the purposes of this document. The methodology should be suitable to support the objectives of the PSA..."

In all of the previous versions of the consultation it stated that the methodology "shall be suitable" to support the objectives of the PSA, but as a suggestion, in the very last round, from only OPG and AECL it was changed to "should be suitable" to support the objectives of the PSA.

My first complaint is that there

is a "should" statement under the requirements section. This is like a "shall should" statement and doesn't make any sense. This is something that the Commission has complained about in the past for licensing.

The rational for the change was that "Under some circumstances, the methodology may be for specific objectives only and this will be noted." This doesn't even make sense. If the objectives, whatever they are, are for a specific reason they will be stated, but they still need to be compliant with them even if they are for a specific reason.

My biggest problems with this change is that the objectives contained in the methodologies are currently part of the licensing basis, and a change from "should" to "shall" actually weakens what is currently in place and would be inconsistent with the *Nuclear Safety and Control Act* and Canada's international obligations under the *Convention of Nuclear Safety*.

The obligations Canada has under the *Convention of Nuclear Safety* is as follows:

> "When necessary in the context of this Convention,

> the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible."

Commissioners, this change will limit your ability to commit to our international obligations. The Commission may find itself having a judicial review with the licensees if it does try to enforce safety improvements if this change is accepted.

It is bad enough that the industry gets to write its own safety goal limits and targets in their methodologies, but making them only guidance is contrary to section 3 of the *Nuclear Safety and Control Act*, which is to provide for the limitation, to a reasonable level and in a manner that is consistent with Canada's international obligations, of the risks to national security, the health and safety of persons and the environment.

Limitation or limit is very well defined legally and is not guidance but a requirement.

Again, I ask the Commission to seek legal guidance to see if the change from shall to should is even within the purview of the Commission because of the legal requirements of the *Nuclear Safety and Control Act* and Canada's international obligations.

To make this change under the premise of a lesson learned from the Fukushima accident is bordering on immoral. It is well understood that the Fukushima accident could have been prevented if the probability of the tsunami had been properly assessed and there was a regulatory requirement to have done something about it.

The CNSC staff are asking you the Commission to make law, the same conditions that caused the Fukushima accident.

One might ask why the CNSC staff would in the last minute make this change. The reality of Canada's Fukushima Action Plan is that after evaluation of the risks posed by nuclear facilities we find that they are not near as safe as we once thought and in most cases do not meet international standards for the prevention of a nuclear accident and require extensive upgrades.

It seems to me that the rules are changing to keep the reactors running, not making them safer. This change shows that the CNSC staff have the same cultural problems as the Japanese regulator.

THE PRESIDENT: Okay. Can you please wind it up?

MR. ROUSE: I'm almost done, sir.

Only, this is much worse because the Canadian regulator should be learning from Japan's mistakes and not following them.

An analogy of the regulatory process in Canada can be made to defence-in-depth. The first defence would be the licensees; the second, CNSC staff assessment of compliance; third, public intervention; and fourth is you, the Commissioners.

Since I have been involved since the Fukushima accident I have seen the public have to rely on the Commission at pretty much every proceeding I have been involved with. The Canadian public should not have to rely on the third and fourth level of defence-in-depth for our safety.

The cultural problems within the Canadian nuclear establishment have been shocking to me. I am very concerned about the results of the recent PIPSC survey.

THE PRESIDENT: Okay. Can you shut him off.

MR. ROUSE: I'm almost done.

THE PRESIDENT: No, you're done now absolutely. Stick to the subject matter. We understand what you are suggesting.

Staff, you want to react to the suggestion on "alignment"?

MR. MOSES: Colin Moses for the record.

I'd just like to clarify, and we got into this a little bit in August.

The Fukushima Task Force said that we should better explain what we're trying to accomplish with a PSA, and so, as a result of that feedback we developed objectives for a PSA. These aren't intended as requirements, they're just a guide to explain what you're trying to accomplish.

And the balance of the document sets out the requirements for meeting those objectives.

And so, that really explains, in section 4.7 when we're saying the methodology should be suitable to meet the objectives, that's what the intent of those objectives are, and so that change that we made was just to clarify what we were hoping to accomplish.

To speak to the suggestion of alignment versus compliance, well, we heard earlier in the *Nuclear Safety and Control Act* that there is an obligation that the Commission prevent unreasonable risk to Canadians and that's where our requirement and our alignment with fundamental safety objective is captured in a regulatory framework, not through a regulatory document on probabilistic safety assessment.

And so, really, the PSA is looking to align with that objective. It's to demonstrate the design is well aligned with that and so it is appropriate to use the term "align" in the document.

THE PRESIDENT: Commissioners, anybody want to jump into this? Comments?

Okay, thank you. Thank you for

the intervention.

I'd like now to open the floor to the Commission. I just want to know if there's any further question on this particular regulatory document, and if there are none, we're done, if there are more, we can take a break and then come back.

You want to finish with this one?

Okay, let me ask the question. Does anybody have any further question on these regulatory documents, these two?

MEMBER VELSHI: I have a quick question.

THE PRESIDENT: Go ahead.

MEMBER VELSHI: I noticed that the small reactor licensees had not submitted any comments on this and I wondered if there was a risk in them not having had a chance to have an input. I know you used your regular process of sending out the document, but would further outreach to them be appropriate?

MR. MOSES: I'd just like to clarify that we did hear from one licensee during our period of feedback on comments in the initial consultation who suggested -- it's AECL, who is operating a small reactor, who suggested that we apply any changes or adjustments made as a result of the feedback on the large reactors equally to the small reactors. So we did receive that feedback.

And as you mentioned, we do push out to an extensive group of stakeholders.

Again, this amendment is changing existing requirements specific to Fukushima, so they're focused and targeted amendments, if that answers your question.

MEMBER VELSHI: So let me ask the question a little differently. Will this document put more requirements on them that they currently don't have?

MR. FRAPPIER: Gerry Frappier for the record.

There's very small reactors in the country and they don't have to do things like PSAs and whatnot. So there is a grading that occurs at the licensing stage with respect to what is needed.

The new reactors that we're getting ready for, if you like, there's nothing that we'd change that would be of significance compared to what we've been talking to in several workshops that we've had with potential vendors.

MEMBER VELSHI: Thank you.

THE PRESIDENT: Monsieur Tolgyesi.

MEMBER TOLGYESI: I just want, on page 2, point 3b, you know, according to the slides that you were discussing, there are two different levels at CNSC and CNSC is actively participating in the development of the third one, third level of PSA. When it will be accepted, developed and accepted, there will be a modification, I suppose, to document 2.4.2 to include the third level.

However, in this b, before last line, we are talking about the first two levels of defence. It gives the impression that there are more than two levels, but right now we have just two.

MR. MOSES: Sir, could you just specify which section you're referring to?

MEMBER TOLGYESI: It's page 2 of document 2.4.2. Page 2, it's in a red one. Point 3, objectives of the probabilistic safety assessment; b, before last line.

We are talking about the first two

levels of defence, but right now we have just two because the third is not developed.

MR. MOSES: Colin Moses for the record. I just want to clarify. There's different levels of PSA but there's multiple levels of defence-in-depth.

MEMBER TOLGYESI: Okay.

MR. MOSES: So we've heard about multiple barriers to protect the public and so that's what we're referring to in that statement.

MEMBER TOLGYESI: Okay. Merci.

THE PRESIDENT: Anything else?

Okay, thank you. We will take a 10-minute break. We'll be back at 4:15. Thank you.

--- Upon recessing at 4:02 p.m. /
Suspension à 16 h 02

--- Upon resuming at 4:16 p.m. /
Reprise à 16 h 16

6.2 Regulatory Document REGDOC 2.5.2, Design of Reactor Facilities: Nuclear Power Plants

THE PRESIDENT: Okay. The next

item on the agenda is their regulatory document REGDOC-2.5.2 for the Design of Reactor Facilities for Nuclear Power Plants as outlined in CMD 14-M18 and CMD 14-M18.A, and I understand that Mr. Frappier will make the presentation. Please go ahead.

CMD 14-M18 / CMD 14-M18.A
Oral presentation by CNSC staff

MR. FRAPPIER: Thank you, Mr. President and Members of the Commission. For the record, my name is Gerry Frappier and I am the Director General of the Directorate of Assessment and Analysis. With me today are Brian Torrie, the Director General of the Regulatory Policy Directorate and Colin Moses, Director of the Regulatory Framework Division. We also have technical and operational staff available to respond to any questions you might have.

We are here today to present REGDOC-2.5.2, the Design of Reactor Facilities for Nuclear Power Plants. Today I will recap the general background on the project and the approach we have taken for public consultation, outline the general feedback we received from our stakeholders, particularly for the second round consultations, and go on to discuss the specific improvements that are proposed for the regulatory document you have before you today.

Finally, we will finish the presentation with a discussion on implementation of these improvements and CNSC staff's conclusions and recommendations.

The purpose of the presentation is to request your approval to publish REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants. You will recall that stakeholders were provided an opportunity to comment on the draft document through a second round of consultations. These comments are being addressed today. We will also address some of the further feedback received in advance of today's meeting.

REGDOC-2.5.2 has been drafted to update requirements for the design of new water cooled nuclear power plants, implement recommendations from the Fukushima Task Force Report and provide guidance to licensees in understanding and complying with the requirements.

This regulatory document is

intended to assist applicants, licensees and reactor vendors in the design, activities and preparation for potential construction of new nuclear power plants. It is also intended to assist CNSC staff with their review of applications to construct new nuclear power plants. Although primarily for new nuclear power plants, this REGDOC will also be used as a modern standard and guidance for the design of nuclear power plants in integrated safety reviews.

CNSC's regulatory document RD-360 contains requirements and guidance for the integrated safety reviews. This includes a review against modern standards to determine reasonable and practical modifications. Upgrades that are safety significant are expected and this document would be used as guidance.

The purpose of REGDOC-2.5.2 is to set out the CNSC set of requirements and guidance for the design of new water cooled nuclear power plants. This regulatory document is risk-informed and aligned with accepted national and international requirements, including the International Atomic Energy Agency's SSR-2/1 Safety of Nuclear Power Plants: Design.

This slide provides an overview of the CNSC document framework and shows that REGDOC-2.5.2 is situated on the Series 2.5: Physical Design. This REGDOC was drafted following the CNSC staff's identification of the need to update existing regulatory document RD-337, which was approved by the Commission and published in 2008. The update was warranted given that RD-337 was written in passive voice using non-mandatory language that made it difficult to identify requirements from guidance. Many of the technical codes and standards in RD-337 were becoming outdated and RD-337 did not contain any guidance to assist licensees in understanding and complying with the requirements. Reviewing RD-337 also provided CNSC staff with the opportunity to implement recommendations from the 2011 Fukushima Task Force Report that pertained to the design of new reactor facilities.

REGDOC-2.5.2, which will supersede RD-337, is based on content from two draft documents which were released for public consultation in 2012. Draft RD-337 Version 2 and GD-337 Guidance for the Design of New Nuclear Power Plants.

Following a CNSC decision on realignment of regulatory documents, the new numbering systems define that RD-337 would become REGDOC-2.5.2. In order to make this document available as soon as possible to stakeholders preparing for potential new-build activities and to help the CNSC meet its commitment to implement Fukushima Task Force recommendations, it was decided to present this document to the Commission in English only.

If REGDOC-2.5.2 is approved by the Commission, the French translation and verification will be ready for publication along with the English version in spring of 2014.

One of the key improvements made to this regulatory document is alignment with the international community. REGDOC-2.5.2 adopts the principles of the recently updated IAEA publication SSR-2/1, Safety of Nuclear Power Plants Design. Key changes adopted from changes to the SSR-2/1 include expansion of the nuclear power plants fundamental safety function to include cooling of all fuels, not just the core, and shielding against radiation. Added requirements for the design to explicitly consider the construction phase, added requirements for fuel to remain usable after anticipated operational occurrences.

In addition, the CNSC conducted a benchmarking study comparing RD-337 Version 1 to regulatory requirements in the U.S., the U.K., France, Finland and the Western European Nuclear Regulators Association. That study found two significant differences.

First, Canadian requirements for electric power systems were limited to only electrical power systems, whereas foreign regulators had more comprehensive requirements in place that included station blackouts and ultimate AC power supplies.

Second, Canadian requirements for operator action times, which is the ability of the nuclear power plant to withstand accident progression without operator action, those action times were less stringent than the international community.

So the CNSC is proposing to align this regulatory document with the international community by strengthening the requirements for electrical power systems. Foreign regulators have more comprehensive electrical power system requirements than those found in the RD-337 and, as a result, the CNSC is proposing to expand the electrical power system requirements in REGDOC-2.5.2 to include normal power supply, standby power supply, emergency power supply and alternate AC power supply.

The CNSC is proposing to amend operator action time so that they are comparable with foreign regulators. The nuclear power plants must remain safe without operator action for a period known as operator action time. Automatic actions and inherent safety characteristics must ensure the safety of the nuclear power plant for this period.

The revised requirements for REGDOC-2.5.2 are more stringent and require NPPs to be able to withstand incidents without operator action for a longer period of time. The change is from 15 to 30 minutes for actions inside the control room and changes from 30 to 60 minutes for actions outside the control room. Most incidents ultimately require operator action and that is what these times our governing. Note that a longer time is allowed for actions outside the control room as operating staff need time to access the location.

A second key improvement to REGDOC-2.5.2 is the implementation of recommendations from the Fukushima Task Force report, including: improved requirement for spent fuel storage; new requirements for portable equipment for use during emergency situations; including redundant connection points to provide water and electrical power in severe accident situations; setting the time for which the plant must be self-sufficient without mobile equipment.

The nuclear power plant must remain safe for eight hours before connection of on-site mobile equipment is needed and 72 hours before off-site services or support are needed. More comprehensive coverage for design extension conditions is also included.

Other key improvements made to REGDOC-2.5.2 include updating terminology to design extension conditions. Design extension conditions, or DECs, are a subset of beyond design-based accidents that are to be considered in the design process of the facility. This is consistent with the original RD-337 done in 2008 which referred to selected beyond design-based accidents. A DEC is an unlikely event and may include fuel melt. An extended loss of AC electric power is an example of a design extension condition. The figure shows how design extension conditions are related to other plant states and conditions. Beyond design-based accidents have no lower frequency limit. DECs are a subset of those beyond design-based accidents that are to be considered in the design; extremely unlikely or physically impossible accidents are not considered in the design. Some DECs may involve severe cord damage, but not all DECs would involve severe core damage.

REGDOC-2.5.2 also updates the way in which complementary design features are classified. Complementary design features are components in the design of nuclear power plants that are necessary to address design extension conditions, they have different design requirements than those features considered for design-based accidents.

REGDOC-2.5.2 requires the availability of fundamental safety functions for design extension conditions. To fulfil the fundamental safety function, complementary design features are provided for design extension conditions. Examples of complementary design features include: provision to cool core debris; provisions to remain in a safe shutdown state to prevent criticality; re-combiners and igniters for hydrogen control; provisions to preclude unfiltered releases; in-vessel retention or core catchers to retain core debris; and, an alternative AC power supply or alternative water supplies beyond what is required in the design basis.

This regulatory document also benefits through the addition of a new section on cyber security to reflect new developments in industrial control systems and current practices in industry, including: transition from analogue to digital technology; wide use of computer-based systems for new designs of nuclear power plants; increased need to protect computer-based systems and equipment from potential cyber-attacks.

This section includes high-level requirements for cyber defensive instrumentation and control architecture and cyber security program. Guidance for meeting the new requirements is also provided.

Finally, REGDOC-2.5.2 assists licensees in understanding and complying with requirements by providing updated and additional references to technical codes and standards, enhanced guidance section, additional information sections which are lists of publications in which additional information may be found. The documents listed in the additional information section provide useful information and guidance.

It should be noted that the selection of design codes and standards is the responsibility of the designer. The CNSC verifies that the set of standards is acceptable.

Moving to public consultations that were done. As noted earlier, REGDOC-2.5.2 is based on two draft documents, RD-337 Version 2, Design of New Nuclear Power Plants, and GD-337, Guidance for the Design of New Nuclear Power Plants. RD-337 Version 2 was issued for public consultation from July 26 to October 4, 2012 and GD-337 was issued for public consultation from August 27 to November 9, 2012.

In total, 138 comments were received on RD-337 and 186 comments were received on GD-337. The stakeholders who commented on the documents included licensees, reactor vendors and international experts.

RD-337 Version 2 and GD-337 were issued for feedback on comments from October 24 to November 8, 2012 and December 14, 2012 to January 8, 2013 respectively. No feedback on comments was received.

REGDOC-2.5.2 you will remember was presented to the Commission for approval on August 22, 2013. In response to a request from stakeholders for additional time to review the document, the Commission asked the CNSC to conduct a further 60-day round of consultations limited to those who participated in the initial consultation. An e-mail invitation to comment on the document was sent to those stakeholders, inviting their comments from August 22 to October 21, 2013.

During this additional round of consultations, the CNSC received 93 comments from three respondents. The revised draft and comment table were sent to stakeholders on January 13, 2014 for their review in advance of today's Commission meeting.

The first key comment the CNSC received on this regulatory document during the second round of consultations is regarding the operator action time. As mentioned, the CNSC is proposing to amend operator action time from 15 minutes to 30 minutes inside the control room and from 30 minutes to one hour outside the control room. These requirements increase the amount of time the design must withstand an accident before which the safety analysis can credit operator actions to halt or mitigate accident progression.

Some reviewers reiterated the request to maintain the current standard of 15 and 30 minutes for inside and outside of the control room respectively. Other reviewers requested that additional information be provided to allow designers to justify alternate action times.

In response to the comments received on this item, the CNSC notes that REGDOC-2.4.1, Deterministic Safety Analysis provides 15 and 30 minutes for existing nuclear power plants and that REGDOC-2.5.2 does not apply to existing nuclear power plants. The CNSC is in agreement that additional guidance related to alternative action time should be provided in the document. To this end, REGDOC-2.5.2 has been revised to include additional information, including a reference to the ANSI standard Time Response Design Criteria for Safety-Related Operator Actions which provides extensive guidance in this area.

The second key comment the CNSC received during the additional round of consultations was regarding structures, systems and components important to safety. In RD-337 Version 1, complementary design features are included in the definition of SSCs Important to Safety. In REGDOC-2.5.2 the CNSC did not propose any changes to this definition.

During the second round of consultations some reviewers stated that including complementary design features in SSCs Important to Safety may extend the meaning of systems important to safety to include beyond design-based accident analysis which may not have the same level of conservatism built into it.

In response to this comment, the CNSC notes that although the probability of structure, systems and components being called upon during design extension conditions is very low, the potential consequences of failures may be large. As a result, it is appropriate to categorize these equipment as important to safety. This does not necessarily predicate a high safety classification.

There is a little bit of complexity here that needs to be explained. After categorizing the equipment, the equipment is then assigned a safety classification. The designer selects an appropriate safety classification using the following factors: (a) the frequency at which the equipment will be called upon to perform a safety function; (b) multiple redundancies to perform a safety function; (c) the time following an event at which the equipment will be called upon to perform a safety function; and, (d) the period for which the equipment will have to perform a safety function.

It should be noted that the safety classification is used to apply appropriate design rules. Furthermore, REGDOC-2.5.2 explicitly states that a low safety class may be appropriate.

As indicated earlier in this presentation, REGDOC-2.5.2 is intended to apply to new nuclear power plants. While the CNSC did not receive any specific comments on the applicability of REGDOC-2.5.2 to existing nuclear power plants, CNSC staff note that in some comments, such as operator action time, reviewers raise the issue of applicability to existing facilities.

To address this, the CNSC notes that the requirements of REGDOC-2.5.2 do not apply to existing nuclear power plants unless they have been included in the licensing basis. Additional text has been added to the preface of the document to clarify this. While REGDOC-2.5.2 is primarily intended to inform the design of new nuclear power plants, it should be noted that existing facilities need to continually demonstrate their safety to the CNSC and adapt their design to meet modern safety requirements where practicable.

In this context, REGDOC-2.5.2 may serve as a tool for licensees of existing facilities in a review against modern codes and standards such as that performed as part of an integrated safety review.

As indicated earlier in this presentation, the revised draft and comment table were sent to the interested stakeholders for review over 60 days in advance of today's

Commission meeting. On February 14 additional comments were received from three stakeholders on this document.

These comments were primarily requesting additional clarification of the CNSC's intent. Some comments were also related to the alignment of the identification of systems important to safety in REGDOC-2.5.2 and in RDGD-98, which is the Reliability Programs for Nuclear Power Plants. A number of additional specific comments and suggestions were received from reviewers and were accommodated by CNSC staff where appropriate.

If approved, REGDOC-2.5.2 will be used by CNSC staff when reviewing application for construction, licences for nuclear power plants or in performing pre-licence vendor design reviews. It will be used to assist stakeholders, including those involved in preparing for potential construction of new nuclear power plants in their design activities and will be published on the CNSC website.

REGDOC-2.5.2 may also serve as a tool for licensees of existing facilities in review against modern codes and standards should such a review be required. REGDOC-2.5.2 will replace RD-337 Design of New Nuclear Power Plants.

In conclusion, overall CNSC staff concludes that REGDOC-2.5.2 Design of Reactor Facilities: Nuclear Power Plants represents a significant improvement in clarifying regulatory expectations for the design of new nuclear power plants. As a result, CNSC staff recommends that the Commission approve REGDOC-2.5.2 for publication and use.

We are now available for any questions you might have. Thank you.

THE PRESIDENT: Thank you. As for the previous regulatory document, I always like to give -- I assume after this exchange of consultation there still may be remnants of disagreements, but I see all the industry and interveners kind of disappear. I assume that is a good thing in this case.

MR. FRAPPIER: Gerry Frappier. Yes, it is a very good thing. I actually during the break went and specifically talked to them and they are all quite okay.

THE PRESIDENT: Okay. So now all you have to convince is us. Okay. So I will start with Dr. McEwan.

--- Laughter / Rires

MEMBER McEWAN: I just have comments and the question. The comment is, to go back to something Mr. Moses says, it will be really helpful to have a common glossary. Having had these three documents to go through and three quite separate and actually quite different glossaries, it would be really helpful to have that and I would suggest that that was a fairly important activity.

THE PRESIDENT: Funny you should mention that, because it has already been assigned as a priority task. They still use the language, "it is under consideration", but I think it is beyond that now, but since you raise it now in public.

MR. FRAPPIER: Gerry Frappier, for the record. Yes, we have a commitment to look at that and to put something together. There are challenges to it.

If you look at the last three documents that we talked about, there is a large breadth of engineering and other disciplines that are being talked about and trying to get them to change the meaning of a word is not always trivial, but we will make it clear at least and we will put a glossary together.

MEMBER McEWAN: That would be very helpful. My second question I think relates to section 11, which is alternative approaches. I had to read this about nine times and I'm still not sure I understand what you mean by this.

Do you mean that if somebody is going to come to you with some alternative technology that is not a water cooled design you will think about what you have to do?

MR. FRAPPIER: Gerry Frappier. Actually, a little bit more precise than that. There are -- we do not want to preclude the potential for innovation in designing reactors and whatnot, so the idea of the section 11 is that if, for a given reason, you cannot meet the specific details of this document, you can -- under section 11 you can demonstrate equivalent levels of safety then we will consider it. So it allows us to have a section where we can consider alternate ways of achieving the same safety that is required here and definitely if it was perhaps not a water cooled reactor, we might have to go to some of those.

MEMBER McEWAN: So let's say it wasn't a water cooled reactor, do you have a process in place that would enable you to evaluate that against this document and against sort of what you might perceive to be new risks introduced by that new technology?

MR. FRAPPIER: Gerry Frappier. Yes, there are two possibilities. So if somebody was planning on introducing into Canada a design that was not a water-based design, or water cooled design, we would certainly be encouraging them to take advantage of our pre-project design review services, if you like, where at that point they could present their safety case. And I think the Commission has been briefed on it a couple of times, so there is a process, there is a methodology that could be used to identify or ensure a) that they understand the requirements that we have; and, b) that they are en route to complying and that we don't see any fundamental barriers to licensing in Canada.

Depending on the design, we may have to do quite a bit of work to find out what in fact our requirements would be for the specifics of a design and this document would be our guide.

THE PRESIDENT: Thank you. Mr. Tolgyesi...?

MEMBRE TOLGYESI : Merci. On page 7.24 you are saying an impact on the industry is that changes to the regulatory framework are not without consequences to the industry. What kind of consequences are you talking?

MR. MOSES: Colin Moses, for the record. Any changes to regulatory requirements can drive some change in the approach and the direction of industry, so we try and be responsive to whether that change is really for a safety benefit or whether it is a necessary burden and it doesn't add any safety value. So that is really what we were trying to get at in that comment.

MEMBER TOLGYESI: Also, this document does not apply to existing nuclear power plants unless they have been included in the licensing basis. How many existing power plants are there where this document is applied, REGDOC-2.5.2?

MR. FRAPPIER: Gerry Frappier. Well, this document is not issued yet, so obviously it doesn't apply to anybody, but if you look at the one before, RD-337 does not apply to any of our operating plants, it is intended to be used for new construction.

MEMBER TOLGYESI: But you are saying that unless they have been included, that means that it could happen.

MR. FRAPPIER: Gerry Frappier. So it could be that if the Commission put it into their licence, then of course it would apply. That would be very, very difficult to do on an operating plant because this is a very detailed design sort of document.

Where we see it useful, though, is when a design is coming up for an integrated safety review that is a requirement before going for refurbishment, for instance, then that process requires a licensee to take his design and to compare it against modern standards, not just design standards, but the safety analyses and all the rest of them and this would be the modern standard that they would compare against and then anyplace there is a gap they need to analyze that gap and demonstrate to us how they are going to get to equivalent levels of safety.

MEMBER TOLGYESI: Did you want to

say something, Mr. Rzentkowski?

DR. RZENTKOWSKI: That is precisely what I wanted to say, that as a part of deciding on the scope of refurbishment activities we implement the standards as a regulatory expectation. So the licensees have to identify the gaps against this regulatory document and disposition those gaps in a risk-informed manner, so that means cost/benefit analysis taken into account.

But it is very important to realize that many of the upgrades implemented as a result of refurbishment projects are actually stemming from the requirements in this document.

MEMBER TOLGYESI: My last is, we are talking about reducing operator's action time from -- increasing from 15 to 30 minutes inside of the control room and to one hour outside, so what is involved technologically and operationally in this decision?

MR. FRAPPIER: Gerry Frappier. Maybe I will change the question just a little bit because, I mean, there are a whole bunch of different systems or things, but I think it's more of -- it's a design constraint, so when the designer is designing the nuclear power plant he's going to have to do all his analyses of what happens under certain circumstances, various accident scenarios and various operational scenarios and in there he has to put down, okay, how much time is it going to take for the operator to do this or that and this provides him with a design restriction, if you like, so that if he needs an operator to do something in 15 minutes or else things might go in a negative safety direction, that would not be allowed, he would have to go to 30 minutes.

THE PRESIDENT: Thank you. Monsieur Harvey...?

MEMBER HARVEY: Two quick questions. Do you expect to receive a new project in the foreseeable future?

MR. FRAPPIER: Gerry Frappier. It is not a Commission decision obviously as to whether we are going to or not. I think from our perspective there has certainly been enough discussion across the country, yes or no, that we better be ready for one, and so what we intend -- what the process here is, is to make sure the regulatory framework is updated to be as modern as possible in case there is one.

And, as Dr. Rzentkowski was saying, also because we know many of our facilities are going to go through refurbishment and we want to compare the requirements against modern standard.

MEMBER HARVEY: Do other countries have a similar document, I mean up to date and equivalent to that one?

MR. FRAPPIER: Gerry Frappier. Most of the major regulators -- not most, all major regulators would have requirements associated with design and, as we mentioned in the presentation, part of what we did in getting ready for this is we went and reviewed in detail the design documents or the design requirements of several other major regulators, just to get a feel for where they are versus where we are.

I know for some of them that are updating, they are now starting to look at ours, so it is one of those things that all the regulators try to keep track of what other people are thinking of and doing.

So yes, they would have a document, it would not be identical to this.

THE PRESIDENT: Thank you. Ms Velshi...?

MEMBER VELSHI: Also a couple of quick questions. So for these last three documents you revised your process a bit and allowed another round of comments. I wondered if you saw value in that and that going forward you are going to change your process for developing regulatory documents to allow stakeholders an opportunity to look at how you disposition their comments and whether they have comments on your disposition?

MR. MOSES: Colin Moses for the record. Yes, that's correct. We have in fact adjusted our practices and our process right now for the post-consultation, so we go through the standard public consultation, invite feedback on any comments received and then we will adjust the document as appropriate.

If we are making substantive changes or if we are changing the approach of a direction as a result of those comments, then we have the option of inviting additional feedback and we will take that option.

The other thing we have done is,

at the very least we recognize that stakeholders need a chance to review how we have responded and how we have addressed their comments, and so we have committed to issuing these documents to all stakeholders who commented at least 60 days ahead of the Commission meeting and that had value both for this document and the previous documents.

We received some additional feedback, minor clarifications in language or intent after issuing those and we have been able to update the documents before presenting them to the Commission.

MEMBER VELSHI: Thank you. And I noticed one of the gaps that you had identified in RD-337 that this document is addressing was a requirement for a design to explicitly consider the construction phase, which is intriguing.

I wonder whether it also looks at the decommissioning phase in the design stage.

MR. FRAPPIER: Gerry Frappier for the record, and I will ask Doug Miller to come and add some commentary on that, please.

MR. MILLER: Doug Miller, for the record. Yes, indeed. When we are looking at the design of the plant at the construction licence stage we are looking at the decommissionability of the plant so that, can the plant be decommissioned easily and other design features for that. We also look at that in the vendor design review process.

MEMBER VELSHI: Yes, but I meant more than doing the review process, does this RD-2.5.2 explicitly ask the designer to make sure they are addressing those considerations in the design?

MR. MILLER: Section 7.24 of REGDOC-2.5.2 cites that requirement.

MEMBER VELSHI: Thank you.

THE PRESIDENT: Thank you. Any other questions? Questions?

You know, I was a bit surprised, but I know that we always talk about ongoing improvement and I actually was happy to see you did a benchmark study.

By the way, is the benchmark study kind of available, posted, or is it one of those people phoning around?

MR. FRAPPIER: Gerry Frappier, I will ask Sang Shim to answer that question.

MR. SHIM: Yes, this is Sang Shim,

for the record. I checked into the website before I came into this Commission hearing. Most of the research contract we post are available on our website, however, this particular contract report only the abstract is available, however, the entire report can be available upon request.

THE PRESIDENT: You know me by now, I like whenever we can share the information, why not post it, particularly if you engage a contractor, so you can actually Hotlink to this particular contractor. I'm not going to go into the detail of how you do this.

But what surprised me here is that, so for example, for the electrical power system, how come we were offside? Offside may be is the wrong word, but not the same as other regulators?

MR. FRAPPIER: I will get -- but just before I turn it over to Sang, I think it is important to clarify that what we are talking about here is the requirement document, not what is actually in the licence and the actual design in the field.

THE PRESIDENT: M'hmm.

MR. FRAPPIER: So certainly all

the facilities in Canada had or have and did have before we did this, electrical systems that had appropriate backups, emergency power systems, seismically qualified, all those sort of things.

What we realized was we didn't have it actually documented that that was a requirement, but perhaps Sang would like to add a bit.

MR. SHIM: This is Sang Shim for the record again. The previous original version had only the section for requirement for emergency power supply only. When we commissioned the research contract benchmarking our requirement against the foreign regulators, one of the two important recommendations they made was our electrical system requirement wasn't comprehensive enough, so we expanded our electrical system requirement section to cover normal standby emergency power supply to alternative power supply requirements. So now our requirements are quite comprehensive and also covers all the aspects of the electrical system designs.

THE PRESIDENT: Kind of my last question is a bit -- I hesitate even to raise it. So here we have a spanking new-build regulatory design, right, so you know a lot of people will ask, will our existing facilities meet some of those design requirements here and how do we reach the bottom line saying that, even though it doesn't apply to existing facilities does not mean our existing facilities are not safe?

MR. FRAPPIER: Gerry Frappier.

THE PRESIDENT: So do we need to explain it somewhere?

MR. FRAPPIER: Gerry Frappier for the record. So we do talk about how this document is going to be used with respect to new -- or current operating facilities in the preface of the document, which is probably about the appropriate level for that document. And I'm going to ask Dr. Rzentkowski perhaps to add a little bit.

But in general, as we said, what we do and what we have continuously done in the CNSC is look at what are the newest standards whenever we are reviewing our operating fleet and then putting the onus on the licensee to demonstrate where he can to close that gap and, if not, what are some alternatives he is going to use, because from an engineering perspective, obviously you can't just come and take a blank --

as if you had a blank piece of paper and redraw your nuclear power plant, you have to work with what you have, but that does not mean that you can't improve it based on new and modern facilities.

As far as how we require that of licensees, perhaps I will ask Dr. Rzentkowski what we are doing now and what maybe we are going to be doing.

DR. RZENTKOWSKI: Thank you, yes. As I indicated in my previous answer, we include this document as modern standard for the review of operating facilities in order to identify the improvements necessary to define the scope of refurbishment activities and, as a result, all operating reactors meet at least the intent of the standard. By "intent", I mean dose acceptance criteria for anticipated operational transience, design basis conditions, and also they are meeting safety goals. So the intent of the standard is being met in a numerical fashion.

In addition, the Fukushima action plan resulted in many additional improvements which again are documented in this particular -- are stemming from this particular document. And those improvements are particularly important for safeguarding spent fuel pools and also safeguarding the plant in the event of extreme external hazard.

So to summarize this long answer, I would say that because of the ISR reviews and because of the Fukushima Action Plan, the safety of operating facilities is very close to the safety which are defined by the requirements in this particular document.

THE PRESIDENT: All I'm saying is you may want to look at your text here to see whether you want -- I know you make reference, but it is really en passant if you may want to use it as a benchmark, but you may want to say something about why -- I understand the application in a refurbishment when it's right to apply new approaches or new on improvement, I just don't see anything here that deals with existing facilities as they are run right now.

It says it doesn't apply to it, but it doesn't necessarily give the assurances that even though it doesn't apply to it, it doesn't mean it's not safe. That was my preoccupation.

MR. FRAPPIER: Gerry Frappier, for the record. I certainly understand what you are proposing and that is why we did modify the preface, although I take your point that it is not comprehensive, but it is an engineering design document.

We have other documents, RD-360, for instance, which talks about how the integrated safety review process is to work and I think that that is looking at being updated as well. So there are other places in our regulatory framework where we make it clear how modern standards are to be used in operating facilities, because it is not just this document, there are also any other standards, we get new fire standards and things like that and how they are going to be incorporated.

THE PRESIDENT: Okay. Anybody? Any other questions? Okay, thank you. Thank you very much. I think this concludes the public meeting of the Commission. Thank you for your patience and have a nice evening.

--- Whereupon the meeting concluded at 4:58 p.m. / La réunion s'est terminée à 16 h 58