FOIA/PA-2011-0357

Group L

RECORDS BEING RELEASED IN THEIR ENTIRETY

Thomas Hipschman

From: Hipschman, Thomas GB5
Sent: Tuesday, August 23, 2011 2:19 PM
To: Batkin, Joshua; Monninger, John
Subject: EQ

Dual unit trip north anna. Loss of off site power, on edg's. Alert declared.

NOUE at salem, hope creek, susq, tmi, calvert and oyster due to feeling eq.

L1

Thomas Hipschman

From: Thomas Hipschman GBJ
Sent: Tuesday, August 23, 2011 5:19 PM
To: Elizabeth Hayden
Cc: Scott Burnell; Joshua Batkin
Subject: Seismic Requirements for North Anna

Per NRR for North Anna

Rock - horizontal acceleration 0.12g
Soil – 0.18g

Vertical is 2/3rds of horizontal

Rock - 0.08g
Soil -0.12g

When I get something for SBO Coping times, I'll let you know.

Tom

Thomas Hipschman
Policy Advisor for Reactors
Office of Chairman Gregory B. Jaczko
301-415-1832

L2

Astwood, Heather

From: Castleman, Patrick
Sent: Wednesday, August 24, 2011 2:23 PM
To: Svinicki, Kristine
Cc: Astwood, Heather; Sharkey, Jeffry; Reddick, Darani
Subject: FW:
Attachments: shakecast_report_nrc.pdf

-----Original Message-----
From: Merzke, Daniel EDO
Sent: Wednesday, August 24, 2011 2:16 PM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike
Cc: Ash, Darren; Brock, Kathryn; McHale, John; Bowman, Gregory; Virgilio, Martin; Hayden, Elizabeth; Powell, Amy
Subject: FW:

There were some questions raised as to whether or not the earthquake exceeded the design basis safe shutdown earthquake for North Anna. The attached comes from USGS, and shows a ground acceleration at North Anna of nearly 0.20g (PGA=19.9918). The SSE for North Anna is 0.12g. The actual number may change, since the actual location and depth of the epicenter seems to fluctuate. As you heard on the call this morning, the licensee is sending their "scratch plates" to California to be analyzed, but there are questions as to the accuracy of the information that can be gleaned from those plates.

At this time NRR/DE is leading the analysis of the event. Region II and NRR have been directed to start communicating with the licensee and stakeholders concerning the path forward in the event that the design basis for the plant has in fact been exceeded. I'll try to keep you updated as additional information is known.

Dan

-----Original Message----- NRR
From: Wilson, George
Sent: Wednesday, August 24, 2011 1:53 PM
To: Wilson, George; Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Brown, Frederick; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary; Merzke, Daniel
Cc: Li, Yong; Karas, Rebecca; Khanna, Meena; Munson, Clifford; Kammerer, Annie; Manoly, Kamal
Subject: RE:

This email contains an update of the ground motion estimate of the North Anna site based on the latest USGS data. The current best estimate of the PGA for the rock is 0.2g, which contains significant uncertainty. The SSE of the North Anna NPP is 0.12g.

It should be noted that the initial estimate from. Version 1 of the ShakeCast report was based on very preliminary information. The present Version 6 is attached. As information has become available,- the ground motion estimate, particularly the location and magnitude of earthquake has become better constrained. This is due to aftershock information and intensity information from the USGS "Did You Feel It?" system, which provides a level of "ground

L3

truthing". We just spoke to the USGS and they think that the numbers have stabilized, however we will provide further updates as we receive them.

The underlying data comes from something called a ShakeMap, which is the information that the USGS puts out to the public. North Anna is on hard rock which may further amplify the incoming motions. It appears that the are many indications that the SSE was exceeded.

Currently, the licensee is retrieving their seismic recording instrumentation. However, we do not yet know the type and quality of the recording data that will be available to the NRC. Information from the NPP will be used to evaluate the USGS estimates of ground motion and to compare with FSAR design basis. The data will also inform the staff if additional analysis is needed.

-----Original Message-----
From: Wilson, George NPP
Sent: Wednesday, August 24, 2011 10:35 AM
To: Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Brown, Frederick; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary
Cc: Li, Yong; Karas, Rebecca; Khanna, Meena; Munson, Clifford; Kammerer, Annie; Manoly, Kamal
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The following attachment includes the following

The latest shake cast report from the USGS - this shows that the peak ground motion was .16g
The Design for North Anna is .12g The North Anna Spectral curve from Numark/FSAR and spectral curve from earthquake. You can see where the data from the earthquake exceeds design curve above 10hz.
The next pictures shows the same curve outlay from the IPEEE submittal..
Reg Guide 1.166 for the after earthquake guidance.

This info is still draft and preliminary.

-----Original Message-----
From: GEORGE.WILSON@NRC.GOV [mailto:george.wilson@nrc.gov]
Sent: Wednesday, August 24, 2011 9:44 AM
To: Wilson, George
Subject:

ED0

U.S.NRC ShakeCast Report USGS science for a changing world

Magnitude 5.8 - VIRGINIA

Version 6

Origin Time: 2011-08-23 17:51:04 GMT

Created: 2011-08-24 15:11:29 GMT

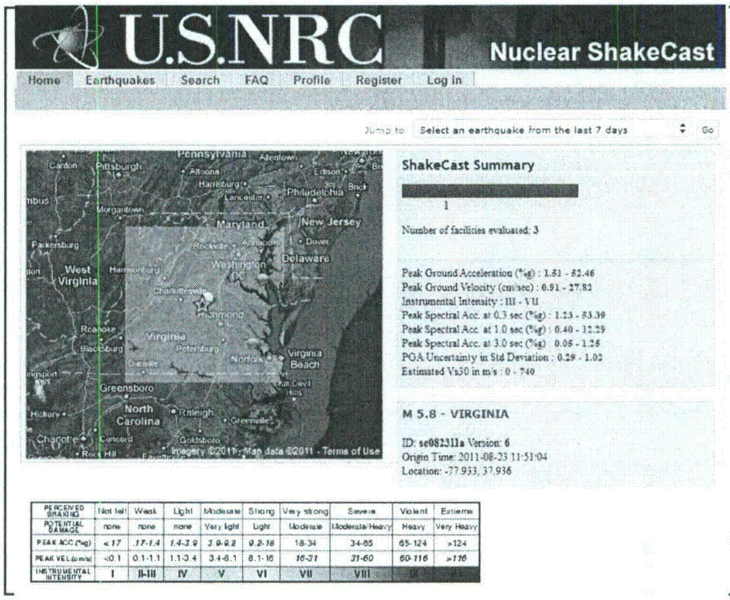
Latitude: 37.9360 Longitude: -77.9330

Depth: 6.0 km

These results are from an automated system and users should consider the preliminary nature of this information when making decisions relating to public safety. ShakeCast results are often updated as additional or more accurate earthquake information is reported or derived.

Map Version 6 Processed Wed Aug 24, 2011 08:50:09 AM MDT -- NOT REVIEWED BY HUMAN

PERCEIVED SHAKING	Not felt	Weak	Light	Moderate	Strong	Very strong	Severe	Violent	Extreme
POTENTIAL DAMAGE	none	none	none	Very light	Light	Moderate	Moderate/Heavy	Heavy	Very Heavy
PEAK ACC.(%g)	<.17	.17-1.4	1.4-3.9	3.9-9.2	9.2-18	18-34	34-65	65-124	>124
PEAK VEL.(cm/s)	<0.1	0.1-1.1	1.1-3.4	3.4-8.1	8.1-16	16-31	31-60	60-116	>116
INSTRUMENTAL INTENSITY	I	II-III	IV	V	VI	VII	VIII	IX	X+



Recent significant earthquakes in the region

- M4.5 VIRGINIA at 12/9/2003 20:59

FACILITY_TYPE	FACILITY_ID	FACILITY_NAME	DIST	LATITUDE	LONGITUDE	DAMAGE_LEVEL	MMI	PGA	PGV	PSA03	PSA10	PSA30
NUCLEAR	USA37	North Anna	18.08	38.0573	-77.7956	YELLOW	VI	19.9918	12.2568	26.0078	5.9443	0.5989
NUCLEAR	USA8	Calvert Cliffs	141.73	38.4319	-76.4424	GREEN	V	6.8436	6.7083	3.5967	1.4285	0.1501
NUCLEAR	USA56	Surry	139.06	37.1633	-76.6942	GREEN	V	6.1296	6.5473	3.5591	1.4118	0.1482

* - MMI level may extend beyond map boundary; some facilities may not appear on the map due to space restriction

Davis, Roger

From: Gilles, Nanette
Sent: Wednesday, August 24, 2011 3:18 PM
To: Apostolakis, George; Baggett, Steven; Davis, Roger; Sosa, Belkys
Subject: Fw:
Attachments: shakecast_report_nrc.pdf

FYI

Sent from my NRC Blackberry

----- Original Message ----- EDO
From: Merzke, Daniel
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike
Cc: Ash, Darren; Brock, Kathryn; McHale, John; Bowman, Gregory; Virgilio, Martin; Hayden, Elizabeth; Powell, Amy
Sent: Wed Aug 24 14:15:33 2011
Subject: FW:

There were some questions raised as to whether or not the earthquake exceeded the design basis safe shutdown earthquake for North Anna. The attached comes from USGS, and shows a ground acceleration at North Anna of nearly 0.20g (PGA=19.9918). The SSE for North Anna is 0.12g. The actual number may change, since the actual location and depth of the epicenter seems to fluctuate. As you heard on the call this morning, the licensee is sending their "scratch plates" to California to be analyzed, but there are questions as to the accuracy of the information that can be gleaned from those plates.

At this time NRR/DE is leading the analysis of the event. Region II and NRR have been directed to start communicating with the licensee and stakeholders concerning the path forward in the event that the design basis for the plant has in fact been exceeded. I'll try to keep you updated as additional information is known.

Dan

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To: Wilson, George; Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Brown, Frederick; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary; Merzke, Daniel
Cc: Li, Yong; Karas, Rebecca; Khanna, Meena; Munson, Clifford; Kammerer, Annie; Manoly, Kamal
Subject: RE:

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L4

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u can see where the data from the earthquake exceeds design curve above 10hz.
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From: GEORGE.WILSON@NRC.GOV [mailto:george.wilson@nrc.gov]
Sent: Wednesday, August 24, 2011 9:44 AM
To: Wilson, George
Subject:

Castleman, Patrick

From: Franovich, Mike
Sent: Wednesday, August 24, 2011 2:46 PM
To: Castleman, Patrick
Cc: Orders, William
Subject: RE:

I think you're correct. This seems to be some type of simulation program that merges recorded data and generates approximations (probably with margins galore).

From: Castleman, Patrick
Sent: Wednesday, August 24, 2011 2:39 PM
To: Franovich, Mike
Cc: Orders, William
Subject: RE:

Thanks, Mike. I kind of thought that was the case. The question I have is whether the USGS value is measured or derived. Based on the map, I'm thinking it was derived.

From: Franovich, Mike
Sent: Wednesday, August 24, 2011 2:30 PM
To: Castleman, Patrick; Merzke, Daniel; Hipschman, Thomas; Marshall, Michael; Gilles, Nanette; Orders, William
Cc: Ash, Darren; Brock, Kathryn; McHale, John; Bowman, Gregory; Virgilio, Martin; Hayden, Elizabeth; Powell, Amy
Subject: RE:

Please note USGS reports the PGA value as percentage of g, so divide by 100.

-----Original Message-----
From: Castleman, Patrick
Sent: Wednesday, August 24, 2011 2:23 PM
To: Merzke, Daniel; Hipschman, Thomas; Marshall, Michael; Gilles, Nanette; Orders, William; Franovich, Mike
Cc: Ash, Darren; Brock, Kathryn; McHale, John; Bowman, Gregory; Virgilio, Martin; Hayden, Elizabeth; Powell, Amy
Subject: RE:

Thanks, Dan. This is timely and helpful.

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L5

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To: Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Brown, Frederick; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary
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Sent: Wednesday, August 24, 2011 9:44 AM
To: Wilson, George
Subject:

Castleman, Patrick

From: Castleman, Patrick
Sent: Thursday, August 25, 2011 8:25 AM
To: Svinicki, Kristine
Cc: Sharkey, Jeffry; Reddick, Darani; Astwood, Heather
Subject: FW: one pager for chairman on North Anna
Attachments: 1 Pager for Chairman Jaczko on North Anna Earthquake Issue.docx

Commissioner,

This came in this morning. It appears that more data and analysis is required before the staff can develop a reasonable understanding of the North Anna design and licensing basis compared to the actual earthquake at the site. Three offices on 18 are keenly interested in how this is evaluated and ultimately resolved. I am concerned that there may be pressure on the staff to jump to unreasonable conclusions, and intend to watch this carefully.

This transmission is also noteworthy because, for the first time in a long time, the staff informed all five Commissioners at the same time. I am not sure whether Merzke acted on his own (he got considerable feedback yesterday from me and some of my colleagues on sharing information) or if he is just following orders.

Pat

From: Merzke, Daniel
Sent: Thursday, August 25, 2011 7:21 AM
To: Monninger, John
Cc: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike; Ash, Darren
Subject: FW: one pager for chairman on North Anna

John, here is a one-pager for the Chairman on the analysis of the seismic activity at North Anna. I hope this "hits the mark." Let me know if there's something else he was looking for.

Dan

From: Wilson, George
Sent: Thursday, August 25, 2011 5:46 AM
To: Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary; Merzke, Daniel; Sanfilippo, Nathan; Hayden, Elizabeth; Chokshi, Nilesh; Wert, Leonard; Hiland, Patrick; Skeen, David
Cc: Li, Yong; Karas, Rebecca; Khanna, Meena; Munson, Clifford; Kammerer, Annie; Manoly, Kamal; Wertz, Trent; Martin, Robert; Thomas, George; Taylor, Robert
Subject: one pager for chairman on North Anna

The attached is the requested one page write up on North Anna from the Chairman's office

George Wilson
USNRC
EICB Branch Chief, Division of Engineering
Mail Stop O12H2
301-415-1711

L-6

Summary of Earthquake Information for the North Anna NPP as of August 24, 2011

The North Anna Nuclear Power Plant (NANPP) has two Safe Shutdown Earthquake (SSE) ground motions, one for structures, systems, and components (SSCs) located on top of rock, which is anchored at 0.12 g, and the other is for SSCs located on top of soil, which is anchored at 0.18 g. The NANPP has two corresponding Operating Basis Earthquake (OBE) ground motion spectra, anchored at 0.09 g for soil and 0.06 g for rock. The figure below shows a comparison between the Safe Shutdown Earthquake (SSE) and OBE for Units 1 and 2, the Unit 3 Combined License (COL) application Ground Motion Response Spectrum (GMRS), the current best estimate of the August 24, 2011 earthquake ground motions from the USGS (ShakeCast version 6), and predicted median and standard deviation earthquake motions using the EPRI ground motion prediction equations. The IPEEE review ground motion (not shown) was anchored at 0.16 g with a similar spectrum as the SSE.

The recent earthquake occurred at a close distance to the plant with a magnitude of 5.8 at a relatively shallow depth. USGS estimates of the maximum ground motion at the plant evolved as new data become available. The current best estimate of the Peak Ground Acceleration (PGA) for the NANPP site is 0.2g, which contains uncertainty and may be updated later. This estimate indicates that the ground motion likely exceeded the SSE response spectra for NANPP Units 1 and 2 (0.12g) over a considerable frequency range, as shown by the green and red points in the figure. The estimated ground motion from the earthquake was not a surprise based on the combined operating license application (COLA) ground motion response spectrum for NANPP Unit 3. This preliminary estimate appears to validate the NRC's current seismic hazard assessment approaches and models for new reactors, as well as the basis for GI-199 reviews.

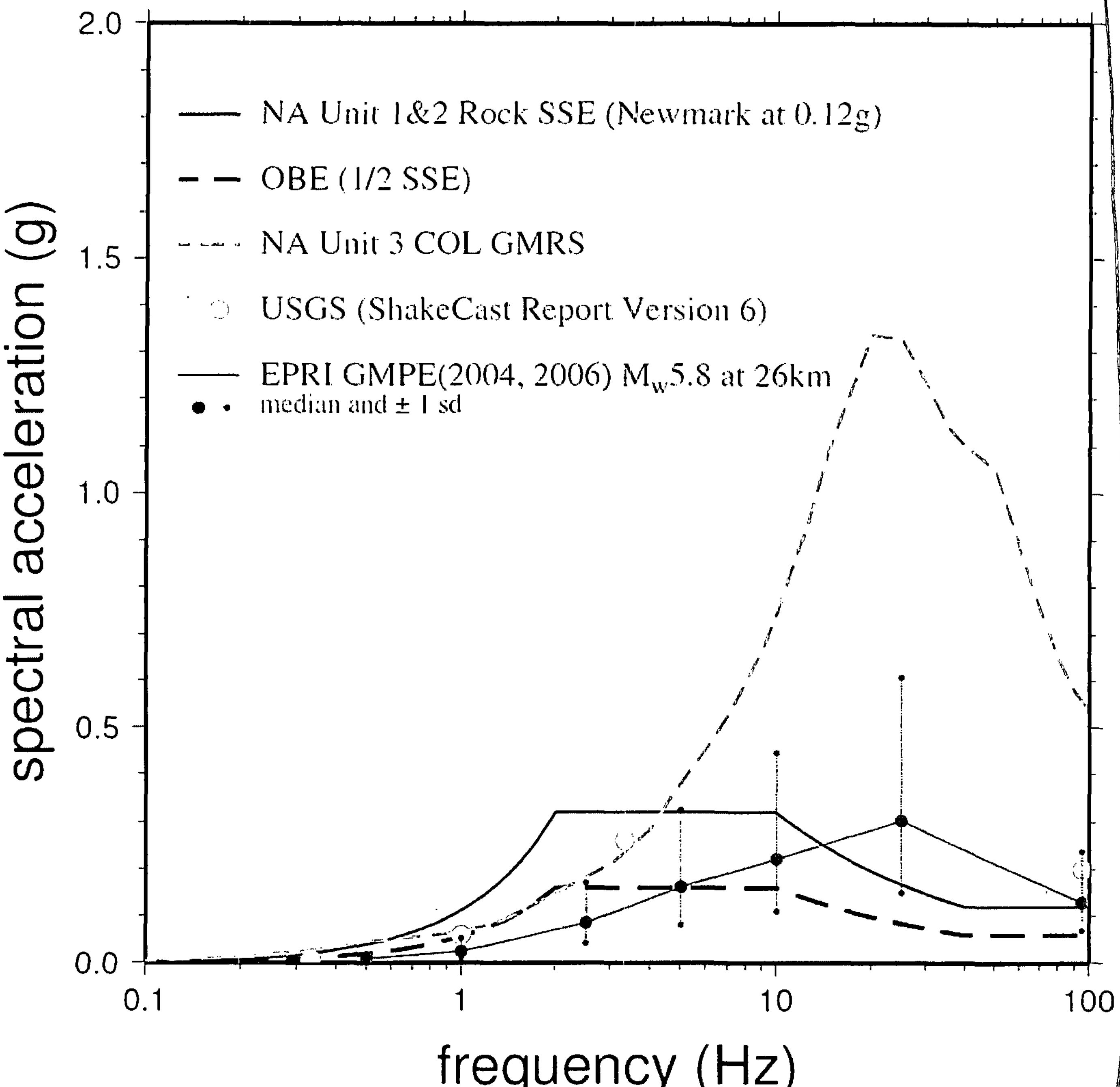
The USGS ground motion estimate values for the plant site are developed based on two types of input. The principal input are theoretical predicted ground motions that come from analyses in which recorded motions at seismograph stations are extended to the NPP sites using ground motion prediction equations (also called attenuation relationships). This theoretical prediction is then modified based on intensity information that comes from the USGS "Did You Feel It?" (DYFI) system. The DYFI system is a method for using large numbers of inputs from affected persons to develop intensity maps that are used as a "ground truth." Currently, the USGS has received nearly 123,000 submitted reports.

NRC staff performed an independent analysis using the best estimate of the earthquake location and magnitude together with the EPRI ground motion prediction equations. The median and ±1 standard deviation curves are shown. It can be seen that the 84th percentile ground motions calculated by the staff are close to the USGS predictions. This makes sense because the USGS theoretical values were increased due to the intensity information that came out of the DYFI system.

Currently, the licensee is retrieving its seismic instrumentation recordings. However, we do not yet know the type and quality of the recording data that will be available to the NRC. Information from the NANPP will be used to evaluate the USGS estimates of ground motion and will be compared against the FSAR design basis. The data will be used to inform the staff whether additional analysis is needed.

The licensee is expected to perform plant walk downs in accordance with RG 1.167, "Restart of a Nuclear Power Plant Shutdown by a Seismic Event," which endorses EPRI's "Guidelines for Nuclear Plant Response to an Earthquake" with conditions. If the SSE is exceeded at certain refer to OEDO

frequencies, the staff will assess the licensee's evaluation of SSCs that are most sensitive to ground motion in that frequency band.



EDO

Apostolakis, George

From:	EDO Update [nrc.announcement@nrc.gov]
Sent:	Monday, August 29, 2011 3:59 PM
To:	Taylor, Renee
Subject:	EDO Update

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

EDO Update

Monday, August 29, 2011

It's been quite a week here in the DC area and the Atlantic Coast as we've dealt aftershocks, as well as a hurricane passing through. While these events are still wanted to address a few related topics. I want to thank all the headquarters an dedication during the past few days and specifically highlight our incident respor and contractors from the Office of Administration (ADM). I thank each of you fo response and teamwork that contributed to the agency's effective handling of th days. I could not be more impressed with the skills, safety focus, and foresight brought to the table. I also recognize that, despite some personal sacrifice, eacl Your performance, dedication and professionalism reflect well on you and the NR

Although a total of 13 Eastern U.S. nuclear power plants declared Unusual Event their sites following the quake, only the North Anna nuclear power plant in Louis Just after 2:00 p.m. on Tuesday, August 23, an Alert was declared at the North station's two units automatically shut down after the facility lost offsite power. E generators provided power to cool the reactors until offsite power was restored a company continues to review data from the quake and assess possible effects. N safety systems has been identified, but Dominion has reported to the NRC that i determined the plant may have exceeded some of the earthquake parameters fc The company and the NRC will continue to carefully evaluate this information to actions may be necessary.

I wanted to mention NRC's new employee notification service, Verizon Notificatic can notify you about office or building closures, weather-related event informati NRC information to the phone numbers and email addresses you designate (sucl personal cell, etc.). ADM used it for the first time this weekend to communicate the headquarters buildings, post-Irene. Employee participation is voluntary, anc enter contact information if you wish to participate. More information about the sign up may be found at:
http://www.internal.nrc.gov/ois/CScatalog/customerservicecatalogs/telecommu also http://www.internal.nrc.gov/announcements/yellow/2011/2011-083.html. I look into it and discuss it with your supervisor.

In terms of our personal safety, the agency has an ***Occupant Emergency Plan*** facility (http://www.internal.nrc.gov/security.html). I urge you to familiarize you

L7

your location. The purpose of the OEP is to reduce the possibility of personal inju
in the event of an emergency. The OEP addresses what to do in the event of fire
winds, electrical power outages, hazardous chemical spills, violent criminal acts,
explosions, medical emergencies, earthquakes, and other conditions. For earthq
OEP directs the following:

- Move away from windows.
- Sit under a sturdy object (such as a desk or table) and hold onto it.
- Be prepared for the aftershocks that may follow an initial earthquake.
- Do not leave the building unless there is a fire or other immediate danger.
- Remember that stairways may be damaged. Exercise extreme care.
- Report any injuries or emergency needs to Security at 415-2000 or call 9-

I know that some staff members had questions such as why other Federal faciliti
following the earthquake while NRC buildings were not. I can only assume that
followed their respective emergency plans just as NRC followed ours. If you hav
OEP, please contact Calvin Byrd.

Thanks in part to the efforts of NRC staff, the Eastern U.S. plants and materials
prepared for Hurricane Irene's impact. Some plants reported emergency sirens
The high winds tore off some aluminum siding and caused a transformer explos
1 that lead to a reactor trip at that plant. NRC staff continues to work with licen
action on Irene's impacts.

Bill

Bill Borchardt, EDO

Thomas Hipschman

From: Hipschman, Thomas GBS
Sent: Thursday, September 01, 2011 6:15 PM
To: Clark, Lisa
Subject: Re: GDC2

Definitely.

----- Original Message -----
From: Clark, Lisa GBS
To: Hipschman, Thomas
Sent: Thu Sep 01 16:25:15 2011
Subject: RE: GDC2

I think I should ask OGC; okay?

-----Original Message-----
From: Hipschman, Thomas GBS
Sent: Thursday, September 01, 2011 3:19 PM
To: Clark, Lisa
Subject: Re: GDC2

Appendix A. Part 50

----- Original Message -----
From: Clark, Lisa GBS
To: Hipschman, Thomas
Sent: Thu Sep 01 15:10:01 2011
Subject: RE: GDC2

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Sent: Thursday, September 01, 2011 3:07 PM
To: Clark, Lisa
Cc: Monninger, John; Coggins, Angela; Batkin, Joshua
Subject: GDC2

Lisa,

The Chairman asked about GDC2 during the North Anna Seismic event briefing.

He would like to know if GDC2 is a one time thing, or would the licensing basis be updated when you have new actual historical data.

Tom

L8

Davis, Roger

From: Gilles, Nanette
Sent: Thursday, September 01, 2011 9:14 AM
To: Apostolakis, George
Cc: Sosa, Belkys; Davis, Roger
Subject: FW: Information on North Anna and Earthquakes

Commissioner – FYI

Nan

Nanette V. Gilles
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-1180
Email: nanette.gilles@nrc.gov

From: Bowman, Gregory EDO
Sent: Thursday, September 01, 2011 7:13 AM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Sosa, Belkys; Gilles, Nanette; Orders, William; Nieh, Ho; Franovich, Mike
Subject: Information on North Anna and Earthquakes

The staff was asked to identify which nuclear power plants had experienced earthquakes exceeding the OBE/SSE. The staff identified two (besides North Anna):

- VC Summer – Experienced an earthquake in 1979 exceeding its OBE. This was prior to the plant being licensed.
- Perry – Experienced an earthquake in 1986 exceeding its OBE and SSE. This was prior to the plant being licensed.

I'm passing this on for information. If you have any questions, please let me know.

Greg

L9

Gilles, Nanette

From: Gilles, Nanette
Sent: Thursday, September 01, 2011 1:43 PM
To: Apostolakis, George
Cc: Sosa, Belkys
Subject: FW: Updated One Pager on North Anna Earthquake
Attachments: One Pager on North Anna Earthquake Issue updated 9-1-11.docx

Commissioner – FYI - Just got this from NRR.

Nan

Nanette V. Gilles
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-1180
Email: nanette.gilles@nrc.gov

From: Bowman, Gregory EDO
Sent: Thursday, September 01, 2011 1:17 PM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Nieh, Ho; Franovich, Mike
Subject: Updated One Pager on North Anna Earthquake

The attached provides a summary of the latest information on North Anna, and includes information related to both the reactors and the ISFSI. We're passing this along for information only. If you have any questions, please let me know.

Greg

L10

EDO | NRR

Summary of Earthquake Information for the North Anna NPP as of August 24, 2011

North Anna Design

The North Anna Nuclear Power Plant (NANPP) has two Safe Shutdown Earthquake (SSE) ground motions, one for structures, systems, and components (SSCs) located on top of rock, which is anchored at 0.12 g, and the other is for SSCs located on top of soil, which is anchored at 0.18 g. The NANPP has two corresponding Operating Basis Earthquake (OBE) ground motion spectra, anchored at 0.09 g for soil and 0.06 g for rock. The figure below shows a comparison between the SSE and OBE for Units 1 and 2, the Unit 3 Combined License (COL) application Ground Motion Response Spectrum (GMRS), the current best estimate of the August 24, 2011 earthquake ground motions from the USGS (ShakeCast version 7), and predicted median and standard deviation earthquake motions using the EPRI ground motion prediction equations.

The current best estimate of the Peak Ground Acceleration (PGA) for the NANPP site is 0.26g, which contains uncertainty and may be updated later. This estimate indicates that the ground motion likely exceeded the SSE response spectra for NANPP Units 1 and 2 (0.12g) over a considerable frequency range, as shown by the green and red points in the figure. The estimated ground motion from the earthquake was not a surprise based on the combined operating license application (COLA) ground motion response spectrum for NANPP Unit 3. This preliminary estimate appears to validate the NRC's current seismic hazard assessment approaches and models for new reactors, as well as the basis for GI-199 reviews.

- The licensee has retrieved its seismic instrumentation recordings from within the plant and has processed the initial information. Preliminary results from the seismic instrumentation indicate some exceedance above the SSE at certain frequencies, depending on the building, measurement direction, and elevation. The information from the NANPP will be used to evaluate the USGS estimates of ground motion and will be compared against the FSAR design basis. The data will be used to inform the staff whether additional analysis is needed.

The licensee has indicated that it will perform plant walk downs in accordance with RG 1.167, "Restart of a Nuclear Power Plant Shutdown by a Seismic Event," which endorses EPRI's "Guidelines for Nuclear Plant Response to an Earthquake" with conditions. If the SSE is exceeded at certain frequencies, the staff will assess the licensee's evaluation of SSCs that are most sensitive to ground motion in that frequency band.

Timeline

- On August 23, 2011, North Anna Power Station declared an Alert due to significant seismic activity onsite from an earthquake which had a measured magnitude of 5.8.

- The licensee conducted the 1st general walkdown of the plant as required by the North Anna Power Station abnormal procedure for seismic event.
- The licensee conducted the 2nd walkdown after the magnitude 4.5 aftershock.
- Preliminary readings of the Seismic Response Spectrum Recorder (scratch plate) and the magnetic tapes identified that the Design Basis Earthquake had been exceeded at certain frequencies. On August 26, the licensee declared all safety-related SSCs of Units 1 and 2 inoperable and issued a 10 CFR 72 Notification

Augmented Inspection Team (AIT)

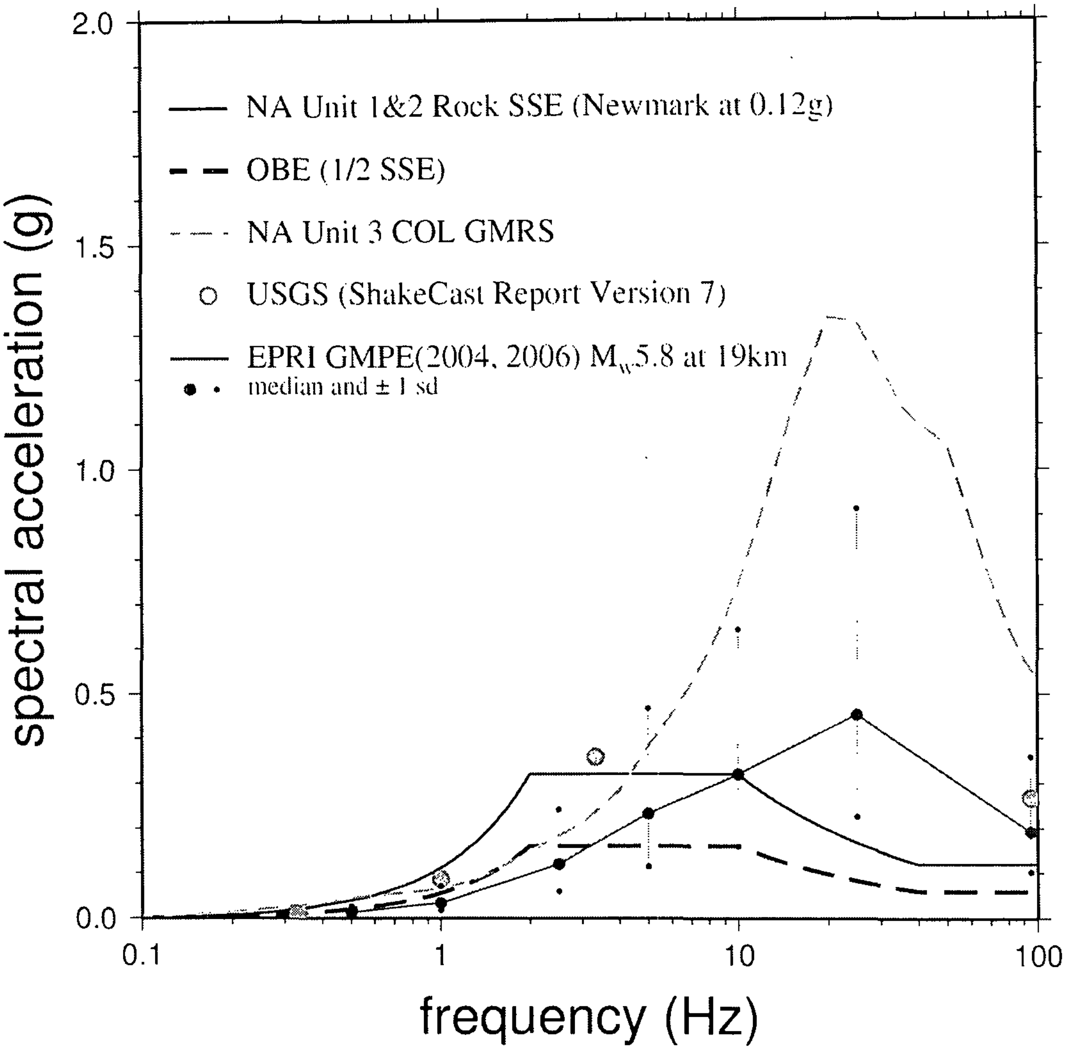
North Anna Earthquake AIT has arrived on site on 8/31/11, held an entrance with the licensee, and commenced the inspection. The data available from the plant instrumentation indicates that the reactor tripped on negative flux rate prior to the loss of offsite power.

- Preliminary raw data provided by the licensee indicate that the DBE has been exceeded in both the horizontal and vertical plane. The licensee is performing a review of the complete data package in order to make a final determination of the ground motion experienced at the site, and expect to have an answer before the end of the week.

- The team has found no indication that any safety related equipment failed during the event, except the 2H EDG. Because of the probability that the DBE has been exceeded the licensee has declared all safety related equipment inoperable and taken action to place the units in a safe condition (CSD).

- Preliminary data on the 2H EDG failure indicates that the cooling water system gasket may have been installed incorrectly.

North Anna Spectrum Curves



EDO|NRR

North Anna Independent Spent Fuel Storage Installation Response to Earthquake

Background:

The North Anna Independent Spent Fuel Storage Installation (ISFSI) uses two spent fuel storage systems manufactured by Transnuclear (TN)

1) Twenty seven vertical TN-32 metal casks under a 10 CFR Part 72 site specific license. This system has a bolted closure lid with a pressure monitoring/alarm system, and stands freely on the ISFSI concrete pad. The design/licensing basis for the vertical TN-32 is controlled primarily by the North Anna ISFSI FSAR and NRC license (SNM-2507) and NRC certificate (1021). The FSAR defines the design acceleration values of 0.18g horizontal and 0.12g vertical, and sliding was not predicted to occur at these values.

2) Twenty six TN NUHOMS HD-32PTH horizontal storage modules (13 loaded) under a 10 CFR Part 72 general license. This system uses a welded-sealed canister and rests on horizontal rails inside the horizontal storage module. The design/licensing basis for the TN NUHOMS HD is controlled primarily by the separate TN-NUHOMS FSAR and NRC certificate (1030), as supplemented by additional site-specific evaluations that were performed by North Anna under 10 CFR 72.212. NUHOMS-HD components are designed to acceleration values of 0.3g horizontal and 0.2g vertical.

Event:

The North Anna ISFSI suffered minor damage from the earthquake:

1) Twenty five of the twenty seven TN-32 casks slid up to 4.5 inches on the concrete pad during the quake. Six cask sets (12 casks) were closer than the 16 foot separation distance specified in the FSAR. There was no damage to the pressure monitors in each cask and no pressure monitoring system alarms during or after the earthquake. There were no crack indications observed in the concrete pad or casks.

2) For the TN-NUHOMS modules, some slight damage was identified around the outlet vents and some surface cracking indications were noted. Additionally, some modules showed gaps between them of approximately 1.5" versus the required 1.0" maximum gap.

Preliminary Determination of Safety Significance:

The staff believes there is no immediate safety issue. The cask designs are robust and consider severe natural phenomena. As expected, the casks withstood the earthquake at North Anna. The spent fuel continues to be surrounded by several tons of steel and concrete, and sealed in an inert helium environment. Damage to concrete components appear to be cosmetic, and does not impact structural integrity or radiation shielding capability. Additionally, the fuel assemblies are designed to withstand a maximum of 4g axial load and 6g lateral load. Inlet and outlet vents were inspected and no exterior blockage was found. Radiation surveys indicate no changes to cask surface dose rates. Thermal performance measurements for all loaded casks found no abnormal temperature differences.

Additionally for the TN-32 casks, the requirement specifying a minimum distance of 16 feet between casks with a heat load greater than 27.1 kW was conservatively established so that the casks do not influence each other thermally and to allow for emplacement on the pad by the cask transporter. Currently, the two casks with the least separation (15 feet, 3.5 inches) are casks that had decay heats of 15.4 kW and 18.0 kW when loaded in 2000 and 2001, both well below the 27.1 kW requirement.

Licensee Response:

The licensee is following RG 1.166, "Pre-Earthquake Planning And Immediate Nuclear Power Plant Operator Post-Earthquake Actions" as a guide to perform their post-event assessment and has completed walkdowns of the ISFSIs

The licensee reviewed this event for reportability under 10 CFR 72.75 (significant reduction in effectiveness of any spent fuel storage cask confinement system) and determined that the TN-32 displacement and NUHOMS-HD damage described above was not reportable.

The licensee contacted TN and provided them with all available pictures, data, and inspection results. TN requested that the licensee perform a more detailed inspection and evaluation of the current condition and sent a team to support this inspection.

NRC Response:

Item 10 of the AIT charter requires the AIT to "Assess the extent of any impact or damage to the Independent Spent Fuel Storage Installation from the seismic event." NMSS and Region II will continue to support the AIT and evaluate information related to the ISFSI to determine whether longer-term licensing or inspection actions are warranted for North Anna or generically.

Thomas Hipschman

From: Hipschman, Thomas GBJ
Sent: Thursday, September 01, 2011 6:15 PM
To: Clark, Lisa
Subject: Re: GDC2

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To:	Apostolakis, George
Cc:	Sosa, Belkys; Davis, Roger
Subject:	FW: Information on North Anna and Earthquakes

Commissioner – FYI

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Greg

L9

North Anna Nuclear Power Plant and Independent Spent Fuel Storage Installation Seismic Event

Presentation to Chairman Jaczko
and Commissioner Apostolakis
September 1, 2011

117

EDO

North Anna Nuclear Power Plant Design Basis

- North Anna Nuclear Power Plant (NANPP) has two Design Basis Earthquake (DBE)* values
  - Structures, systems, and components (SSCs) founded on top of rock anchored at 0.12 g and SSCs founded on top of soil anchored at 0.18 g
- NANPP has two corresponding Operating Basis Earthquake (OBE) values, anchored at 0.06 g for rock and 0.09 g for soil (OBE is ½ of the DBE)

* Design Basis Earthquake means the same as Safe Shutdown Earthquake

Sequence of Events

- On August 23, 2011, North Anna Power Station declared an Alert due to significant seismic activity onsite from an earthquake which had a measured magnitude of 5.8.
- The licensee conducted the 1st general walkdown of the plant as required by the North Anna Power Station abnormal procedure for seismic event.
- The licensee conducted the 2nd walkdown after the magnitude 4.5 aftershock.
- Reactor Seismic Response Spectrum Recorder (scratch plate) readings identified that the Design Basis Earthquake had been exceeded at certain frequencies.
- On August 26, the licensee declared all safety-related SSCs of Units 1 and 2 inoperable and issued a 10 CFR 50.72 Notification

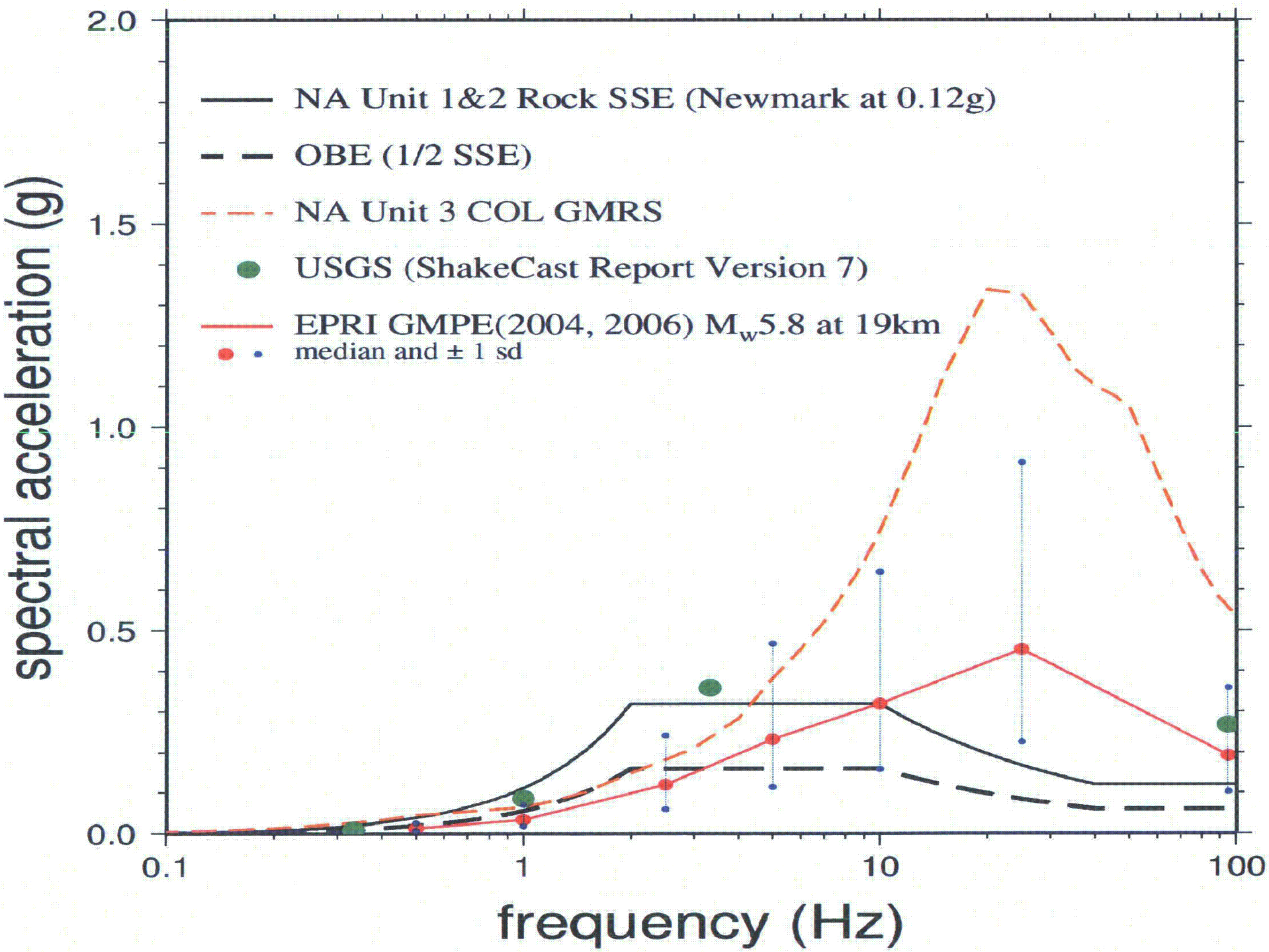
History Of Seismic Events

- Exceedance of the DBE is an unprecedented event at an operating unit or ISFSI
- While Perry Unit was under construction, an earthquake occurred that exceeded SSE at high frequency (15hz)
  - A special safety inspection was conducted by the NRC's Region III Staff on February 5–7, 1986. *See* Inspection Reports 50–440/86005 and 50–440/86006. This included a post-earthquake walkdown and visual inspection of an extensive list of safety-related systems and components.

Regulatory Requirements for Restart of Reactors

- Appendix A to Part 100—Paragraph V(a)(2) states, "*If vibratory ground motion exceeding that of the Operating Basis Earthquake occurs, shutdown of the nuclear power plant will be required.*
  - *Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage occurred to those features necessary for continued operation without undue risk to the health and safety of the public."*
- *10 CFR 50.54 (ff) contains similar language for Appendix S plants. (Appendix S applies to Part 52 applicants and operating reactor construction permits submitted on or after Jan. 10, 1997)*
- Director of NRR will authorize restart

Staff Initial Assessment



Licensee Initial Assessments

- Licensee's evaluation of reactor seismic instruments indicate that the SSE was exceeded at some frequencies.
- Information from NANPP's seismic recordings will be utilized by the staff to assist in the assessment of the licensee's operability determination.

North Anna Independent Spent Fuel Storage Installation Design Basis

- The North Anna Independent Spent Fuel Storage Installation (ISFSI) uses two spent fuel storage systems manufactured by Transnuclear (TN):
  - Vertical TN-32 metal casks under a 10 CFR Part 72 site specific license. The FSAR defines the design acceleration values of 0.18g horizontal and 0.12g vertical, and sliding was not predicted to occur at these values.
  - Horizontal TN-NUHOMS under their general license. NUHOMS-HD components are designed to acceleration values of 0.3g horizontal and 0.2g vertical.
  - The fuel assemblies are designed to withstand a maximum of 4g axial load and 6g lateral load.

North Anna ISFSI Earthquake Impacts

- TN-32:
  - Several casks slid up to 4.5 inches - six cask sets (12 casks) were closer than the 16 foot separation distance specified in the FSAR
  - No damage to the pressure monitors in each cask and no pressure monitoring system alarms during or after the earthquake
  - No crack indications observed in the concrete pad or casks
- TN-NUHOMS:
  - Slight damage around the outlet vents and some surface cracking indications identified
  - Some modules showed gaps between them of approximately 1.5″ versus the required 1.0″ maximum gap

Staff Preliminary Assessment of ISFSI Damage

- The staff believes there is no immediate safety issue:
  - The spent fuel continues to be surrounded by several tons of steel and concrete, and sealed in an inert helium environment
  - Damage to concrete components appear to be cosmetic, and does not impact structural integrity or radiation shielding capability
  - Inlet and outlet vents were inspected and no exterior blockage was found
  - Radiation surveys indicate no changes to cask surface dose rates
  - Thermal performance measurements for all loaded casks found no abnormal temperature differences

Augmented Inspection Team

- AIT was dispatched on August 31, 2011, which will be conducted in accordance with MD 8.3, "NRC Incident Investigation Program."
- Objectives of the AIT include:
  - Collect, analyze and document factual information and evidence
  - Assess licensee's actions and plant equipment response during the earthquake and aftershocks
  - Assess the extent of any impact or damage to the ISFSI from the seismic event
  - Conduct independent extent of condition review
  - Collect information to support final determination of risk significance of event
  - Identify generic issues associated with the event

Apostolakis, George

From: Gilles, Nanette
Sent: Tuesday, September 06, 2011 9:49 AM
To: Apostolakis, George
Cc: Sosa, Belkys
Subject: FW: FYI - Information from North Anna Briefing
Attachments: Safety and Risk Assessment for GI-199.pdf

Commissioner – I have added the safety assessment report for GI-199, and all of the appendices, to the G drive, GI 199 folder.

Nan

Nanette V. Gilles
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-1180
Email: nanette.gilles@nrc.gov

From: Bowman, Gregory OEDO
Sent: Thursday, September 01, 2011 4:09 PM
To: Gilles, Nanette
Subject: FYI - Information from North Anna Briefing

Nan,

At today's briefing on North Anna and GI-199, Commissioner Apostolakis asked for the safety/risk assessment for GI-199. I attached a copy.

The safety/risk assessment is an enclosure to a memo discussing the GI. There are also four appendices with additional supporting information. The package can be found at ML100270582 – it gives you the memo, the enclosure, and the appendices.

You can also find a lot of information related to the GI on RES's website at the following location:

http://www.internal.nrc.gov/RES/projects/GIP/Individual%20GIs/GI-0199.html

If you need anything else, please let me know.

Greg

L12

public

GENERIC ISSUE 199 (GI-199)

IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES IN CENTRAL AND EASTERN UNITED STATES ON EXISTING PLANTS

SAFETY/RISK ASSESSMENT

August 2010

TABLE OF CONTENTS

EXECUTIVE SUMMARY

RES staff developed and implemented a methodology to assess the risk associated with this issue. Preliminary results indicate that the issue should continue to the Regulatory Analysis Stage of the Generic Issues Program (GIP) for further investigation to identify candidate backfits and evaluate their potential cost-justified imposition. The information needed to perform the Regulatory Assessment is not currently available to the staff. The methodology, analyses, results and limitations of the safety risk assessment are summarized below.

Risk Methodology

Seismic core damage frequency (SCDF) was chosen as the appropriate risk metric because it is expected to be more sensitive than other metrics (either large-early release fraction or public dose) to changes in the seismic hazard. In addition, SCDF can be estimated using Individual Plant Examination of External Events (IPEEE) information. Conversely, the IPEEE program did not produce sufficient quantitative information to perform estimation of alternate risk metrics.

The staff performed a two-stage assessment to determine the implications of updated probabilistic seismic hazards in the Central and Eastern U.S. (CEUS) on existing nuclear power plants (NPPs). The change in seismic hazard with respect to previous estimates at individual NPPs was evaluated in the first stage, and the change in SCDF as a result of the change in the seismic hazard for each operating NPP was estimated in the second stage. The seismic hazard at each NPP site is dependent on the unique seismology and geology surrounding the site which necessitated separately determining the implications of updated probabilistic seismic hazard for each of the 96 operating NPPs in the CEUS.

Approximate SCDF estimates were developed using a method which includes integrating the mean seismic hazard curve and the mean plant-level fragility curve for each NPP. This method, developed by Kennedy (1997), is discussed in Section 10.8.9 of AMSE/ANS RA-Sa-2009 and has previously been used by the staff in the resolution of GI-194, "Implications of Updated Probabilistic Seismic Hazard Estimates," and during reviews of various risk-informed license amendments. This approach was discussed with EPRI under an NRC-EPRI seismic research memorandum of understanding. EPRI agreed that this is a reasonable approach for evaluating GI-199.

Performance of the Safety/Risk Assessment

The following describes the details of performing the Safety/Risk Assessment and associated limitations. There are two discrete inputs required for the methodology described above, plant-specific seismic hazard information and estimates of plant-specific seismic fragility.

Seismic Hazard Curves

SCDF estimates were produced using three sets of mean seismic hazard curves representing a range of different assumptions and the changing state of knowledge:

- EPRI, 1989
- Lawrence Livermore National Laboratory (LLNL), 1994
- NRC based on U.S. Geological Survey (USGS), 2008

Plant-Level Fragility Curves

Plant-level fragility curves were developed from information provided in the IPEEE submittals. About one-third of the plants in the CEUS performed a seismic probabilistic risk assessment (SPRA) as part of their IPEEE program. About two-thirds of the SPRA plants provided plant-level fragility information (either in tabular or graphic format) in their IPEEE submittals. The remaining one-third of the SPRA plants provided SCDF estimates based on a variety of seismic hazard curves (EPRI 1989, LLNL1994, or site-specific curves developed specifically for the IPEEE program). For these remaining plants, plant-level fragility values were back-calculated by matching the reported SCDFs and using engineering judgment. In cases where reasonable engineering judgments could not be readily made, sensitivity studies were performed.

The other two thirds of the plants conducted a seismic margins analysis (SMA) as part of their IPEEE program. The figure of merit for an SMA is the plant-level high confidence of low probability of failure (HCLPF) value.

Analyses Performed

For each of the three sets of seismic hazard curves (EPRI, LLNL, NRC/USGS), four SCDF estimates were developed. These four SCDF estimates were developed for a discrete series of representative spectral response frequencies (peak ground acceleration (PGA), 10, 5, and 1-Hz) and utilized spectral shapes based on the plant-specific IPEEE evaluations. For each NPP and hazard curve combination, the discrete spectral SCDF estimates were combined using four different weighting schemes to produce final plant-level SCDF estimates.

Evaluation of Changes in Seismic Hazard Estimates

The evaluation of the potential significance of changes in seismic hazards was performed in a stepwise fashion by posing a series of questions that indicated the degree of deviation of seismic hazard estimates developed using the most recent seismic hazard information and staff guidance from previously developed assessments. The previous assessments included the Safe Shutdown Earthquake (SSE), the review level earthquake (RLE) used in the IPEEE assessment, and the 1989 EPRI and 1994 LLNL seismic hazard studies. The comparison of results indicated a substantial increase in the estimated seismic hazard values relative to all previous assessments for a number of plants.

Risk Results

For those plants with increases in seismic hazard estimates, the study next evaluated if there was any significant change in the risk metric (SCDF). To perform this assessment, the point estimates of the mean SCDF developed using the NRC/USGS hazard curves were compared with the baseline SCDFs developed using the original LLNL or EPRI seismic hazard curves. The SCDF changes for a number of plants lie in the range of 10-4/year to 10-5/year, which meet the numerical risk criteria for an issue to proceed to the GIP Regulatory Assessment Stage.

Overall seismic risk estimates remain small in an absolute sense. All operating plants in the CEUS have seismic core-damage frequency (SCDF) less than or equal to 10^{-4}/year, confirming that there is no immediate concern regarding adequate protection.

Limitations of the Risk Methodology and Data Used

The approach used to estimate SCDF in the Safety/Risk Assessment is highly sensitive to the inputs used. While work to date supports a decision to continue to the GIP Regulatory Assessment Stage; the methodology, input assumptions, and data are not sufficiently developed to support other regulatory decisions or actions.

The approach used to estimate SCDF in the Safety/Risk Assessment does not provide insight into which structures, systems, and components (SSCs) are important to seismic risk. Such knowledge provides the basis for postulating plant backfits and conducting a value-impact analysis of potential backfits during a regulatory analysis.

Little useful information exists regarding plant seismic capacity (the ability of a plant's SSCs to successfully withstand an earthquake) beyond the required design-basis level for a number of plants that performed reduced-scope SMAs.

In general, only limited, qualitative information about the seismic capability of containments is provided in IPEEE submittals.

The integration of the mean seismic hazard curve and the mean plant-level fragility curve is not equal to the mean SCDF; accordingly, the SCDF estimates produced by the approach are point estimates.

The approach does not provide a quantitative estimate of the parametric uncertainty in the SCDF. Although the USGS approach explicitly includes uncertainties, the USGS has not published fractile curves for its seismic hazard estimates.

New consensus seismic hazard estimates for the CEUS will become available in late 2010 or early 2011 (these are a product of a joint NRC, Department of Energy, USGS, and EPRI project), and underscore the need to develop a regulatory mechanism to routinely and promptly evaluate new seismic hazard information as it becomes available.

Problems that currently exist with producing realistic SCDF estimates will continue even after the new consensus seismic hazard estimates are developed. The main problem is that many IPEEEs did not produce SCDF estimates and so lack some of the information needed to produce updated SCDF estimates. As such, the available seismic margins can only be grossly estimated and may be eroding as new seismic hazard estimates are developed.

<u>Information Needed to Perform the Regulatory Analysis Stage of GI-199</u>

The following four categories of information are needed to perform the Regulatory Analysis Stage of the GIP for GI-199:

- *Site-specific, updated EPRI hazard curves* used to evaluate plant seismic risk in the recent study conducted by EPRI for industry. The hazard curves should cover a range of appropriate structural frequencies (PGA to 0.5 Hz), and be in a tabular, digital form.

- *Frequency dependent, site-specific amplification functions* used to translate seismic motions from hard rock conditions to appropriate surface conditions. These functions should be consistent with the recent seismic evaluation performed by EPRI using updated seismic hazard results (see previous item), and be in tabular, digital form.

- *Plant-level fragility information* used in the recent study conducted by EPRI. Specific information needed includes the median seismic capacity (C_{50}), the composite logarithmic standard deviation (β_C), and spectral ratios (relative to PGA) for 1, 5, and 10 Hz (at a minimum), representative of the currently operated plant.

- *Plant-specific significant contributors to seismic risk.* Identify the SSCs that are significant contributors to seismic risk and the approach used to identify them.

Conclusions

Results of the Safety/Risk Assessment indicate that there is no immediate concern regarding adequate protection, but that the issue should continue to the Regulatory Analysis Stage of the GIP (for further investigation regarding possible cost-justified backfits). The information and methods needed to perform the Regulatory Assessment are not yet available to the staff, but have been identified.

LIST OF ACRONYMS AND INITIALISMS

AEF	annual exceedance frequency
CAV	cumulative absolute velocity
CDF	core-damage frequency
CEUS	Central and Eastern United States
COL	Combined License
EPRI	Electric Power Research Institute
EPRI-SOG	Electric Power Research Institute-Seismicity Owners Group
ESP	Early Site Permit
FSAR	Final Safety Analysis Report
GIP	Generic Issues Program
GMRS	ground motion response spectrum
HCLPF	high confidence of low probability of failure
HR	hard rock
IPEEE	Individual Plant Examination of External Events
LERF	large early-release frequency
LLNL	Lawrence Livermore National Laboratory
MD	Management Directive
MOU	Memorandum of Understanding
NPP	nuclear power plant
NRR	Office of Nuclear Reactor Regulation
PGA	peak ground acceleration
PSHA	probabilistic seismic hazard analysis
RLE	review-level earthquake
SA	spectral acceleration
SCDF	seismic core-damage frequency
SCDOT	South Carolina Department of Transportation
SMA	seismic margins analysis
SPRA	seismic probabilistic risk analysis
SR	soft rock
SSC	structures, systems, and components
SSE	safe shutdown earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
TFI	Technical Facilitator Integrator
TIP	Trial Implementation Program
TVA	Tennessee Valley Authority
UHS	uniform hazard spectrum
USGS	United States Geological Survey
Vs	shear wave velocity
WUS	Western United States

GLOSSARY

- *Annual exceedance frequency (AEF)* – Expected number of occurrences per year where a site's ground motion exceeds a specified acceleration.

- *Design basis earthquake* or *safe shutdown earthquake (SSE)* – A design basis earthquake is a commonly employed term for the SSE: that earthquake for which certain structures, systems and components are designed to remain functional. In the past, the SSE has been commonly characterized by a standardized spectral shape anchored to a "peak ground acceleration" value.

- *Ground acceleration* – Acceleration at the ground surface produced by seismic waves, typically expressed in unit of g, the acceleration of gravity at the Earth's surface.

- *High confidence of low probability of failure (HCLPF) capacity* – A measure of seismic margin. In seismic risk assessment, this is defined as the earthquake motion level at which there is a high confidence (95%) of a low probability (at most 5%) of failure.

- *Seismic hazard* – Any physical phenomenon, such as ground motion or ground failure, that is associated with an earthquake and may produce adverse effects on human activities (such as posing a risk to a nuclear facility).

- *Seismic margin* – The difference between a plant's HCLPF capacity and its seismic design basis (safe shutdown earthquake, SSE).

- *Seismic risk* – The risk (frequency of occurrence multiplied by its consequence) of severe accidents at a nuclear power plant that are initiated by earthquakes. A severe accident is an accident that causes core damage and, possibly, a subsequent release of radioactive materials to the environment. Several risk metrics may be used to express seismic risk, such as seismic core-damage frequency and seismic large early release frequency.

GENERIC ISSUE 199 (GI-199)
IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES IN CENTRAL AND EASTERN UNITED STATES ON EXISTING PLANTS

SAFETY/RISK ASSESSMENT

1. BACKGROUND

In support of early site permits for new reactors, the U.S. Nuclear Regulatory Commission (NRC) staff reviewed updates to seismic source and ground motion models provided by applicants. The seismic update information included new models to estimate earthquake ground motion and updated models for earthquake sources in seismic regions around Charleston, South Carolina, New Madrid, Missouri, and southern Illinois and Indiana. The new data and models resulted in increased estimates of the seismic hazards at many plants in the Central and Eastern United States (CEUS), but these estimates remain small in an absolute sense. The staff reviewed and evaluated this new information along with recent U.S. Geological Survey (USGS) seismic hazard estimates for the CEUS. From this review, the staff identified that the estimated seismic hazard levels at some current CEUS operating sites might be higher than seismic hazard values used in design and previous evaluations. Figure 1 shows a comparison of response spectral values based on Early Site Permit (ESP) seismic hazard results with those previously developed as part of the 1989 Electric Power Research Institute-Seismicity Owners Group (EPRI-SOG) study for an annual exceedance frequency (AEF) of 10^{-5}. The figure shows that for four of the ESP submittals (North Anna, Grand Gulf, Vogtle, and Clinton), the seismic hazard is higher over most of the frequency range compared to the earlier EPRI-SOG study results.

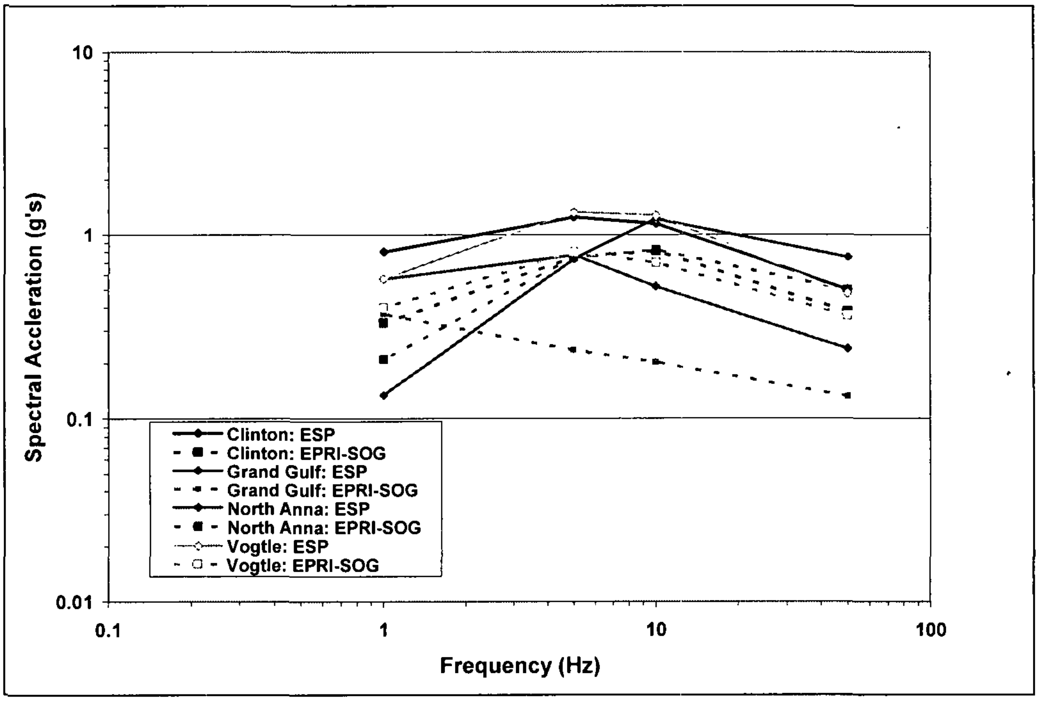


Figure 1. Comparison of Seismic Hazard Results for Four Early Site Permit Submittals (Solid Lines) to 1989 EPRI-SOG Results (Dashed Lines). Curves are response spectral values (5-percent damping) at an annual exceedance frequency of 10^{-5}.

The staff of NRC's Office of Nuclear Reactor Regulation (NRR) compared the new seismic hazard data with the earlier evaluations conducted as part of the Individual Plant Examination of External Events (IPEEE) Program. From this comparison, the staff determined that seismic designs of operating plants in the CEUS still provide adequate safety margins; however, the staff continues to evaluate new seismic hazard data and models and their potential impact on plant risk estimates. At the same time, the staff also recognized that the new seismic data and models could reduce available safety margins because of increased estimates of the probability associated with seismic hazards at some of the currently operating sites in the CEUS. The licensing basis for these plants does not include a probabilistic assessment of seismic hazards or a probabilistic assessment of their potential impact on plant structures, systems, and components (SSCs). Rather, the licensing basis for these plants is based on deterministic analysis for design basis loads from the maximum earthquake level that is determined from historical data (10 CFR 100 Appendix A). On May 26, 2005, the NRR staff issued a memorandum (ADAMS Accession No. ML051450456) recommending that the new data and models on CEUS seismic hazards be examined using a probabilistic approach under the Generic Issues Program (GIP) to help assess the potential reduction in available safety margins.

The staff completed a screening analysis using guidance contained in Management Directive (MD) 6.4 and SECY-07-0022 in December 2007 and reconvened the screening panel in January 2008. On February 1, 2008, the RES Director approved the screening panel's recommendation (ADAMS Accession No. ML073400477) to begin the Safety/Risk Assessment Stage of the Generic Issue Process. On February 6, 2008, the staff met with the public and stakeholders to discuss the results of the Screening Stage of Generic Issue 199.

On March 14, 2007, NRC and the Electric Power Research Institute (EPRI) signed a Memorandum of Understanding (MOU) on Cooperative Nuclear Safety Research. On July 11, 2008, NRC signed an addendum to this MOU concerning seismic risk, and on July 22, 2008, EPRI also signed the addendum. Program Element 3A of this addendum addresses updated seismic hazard assessments in support of GI-199. NRC and EPRI met on December 3, 2008, in Ft. Lauderdale, Florida, and March 17-18, 2009, in Palo Alto, California, to exchange information about seismic risk methodology, seismic hazard curves, and current seismic core-damage frequency estimates for operating nuclear power plants (NPPs). Under the terms of the MOU, data acquired during the course of collaborative work are considered privileged information and thus are routinely withheld from release until the final reports on this work are made publically available.

2. PURPOSE

The purpose of the Safety/Risk Assessment Stage is twofold:

- Determine, on a generic basis, if the risk associated with Generic Issue (GI) 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States (CEUS) on Existing Plants," warrants further investigation for potential imposition as a cost-justified back-fit.
- Provide a recommendation regarding the next step (i.e., should the issue continue to the Regulatory Assessment Stage for identification and evaluation of potential generic, cost-justified backfits, be dropped due to low risk, or have other actions taken outside the Generic Issues Program).

3. APPROACH

To determine the implications of updated probabilistic seismic hazard estimates in the CEUS on existing NPPs, the staff performed a two-stage assessment. One stage involved evaluating the change in seismic hazard with respect to previous estimates at individual NPPs (discussed in section 4.2). The second stage estimated the change in seismic core-damage frequency (SCDF) as a result of the change in the seismic hazard for each operating NPP in the CEUS (discussed in section 4.1). This approach was based on the following considerations:

- The estimation of seismic hazards is complex and significant uncertainties are associated with many of the input parameters in the hazard models. This is especially true for regions of lower seismic activity such as the CEUS. Evaluation of any new seismic hazard estimates with respect to previous estimates is prudent to ensure the changes are significant and not merely representative of the fidelity in the seismic hazard estimation process.
- MD 6.4 states that the risk-informed technical assessment of a generic issue may be conducted using core-damage frequency (CDF), large early-release frequency (LERF),

public dose (person-rem), or a combination of these risk metrics. The selection of the appropriate risk metric(s) to assess a generic issue depends on the specific nature of the generic issue being assessed. The Safety/Risk Assessment of GI-199 involves the implications of updated probabilistic seismic hazard estimates that describe the distribution (frequency and size) of seismically induced site vibratory ground motions at NPP sites. Although each of the three risk metrics (CDF, LERF, and public dose) depends on the seismic hazard, SCDF is expected to be the most sensitive to changes in the seismic hazard.

- Given a limited number of assumptions, SCDF can be readily estimated using the seismic hazard and information from the IPEEE requested by Generic Letter 88-20, Supplement 4. In contrast, the containment performance analyses conducted under the IPEEE program did not produce sufficient quantitative information to allow the estimation of either LERF or public dose.

- Typically, the Safety/Risk Assessment of a generic issue is based upon a surrogate probabilistic risk assessment (PRA) or a small set of surrogate PRAs that model classes of plants (e.g., four-loop Westinghouse pressurized-water reactors [PWRs], Babcock and Wilcox PWRs, boiling-water reactors [BWRs], etc.). However, the seismic hazard at each NPP site is unique because it depends on the seismology and the geology surrounding the site. Figure 2 illustrates this point and illustrates the large variation in the seismic hazard across the United States. The Safety/Risk Assessment needed to determine the extent of GI-199 (e.g., determine how many plants are potentially affected). Therefore, it was necessary to determine the implications of updated probabilistic seismic hazard estimates in the CEUS at each operating NPP.

With respect to the Safety/Risk Assessment Stage, the term "Central and Eastern United States" refers to operating NPPs that are located east of the Rocky Mountains. Table 1 lists the 96 operating NPPs that are located within the CEUS.

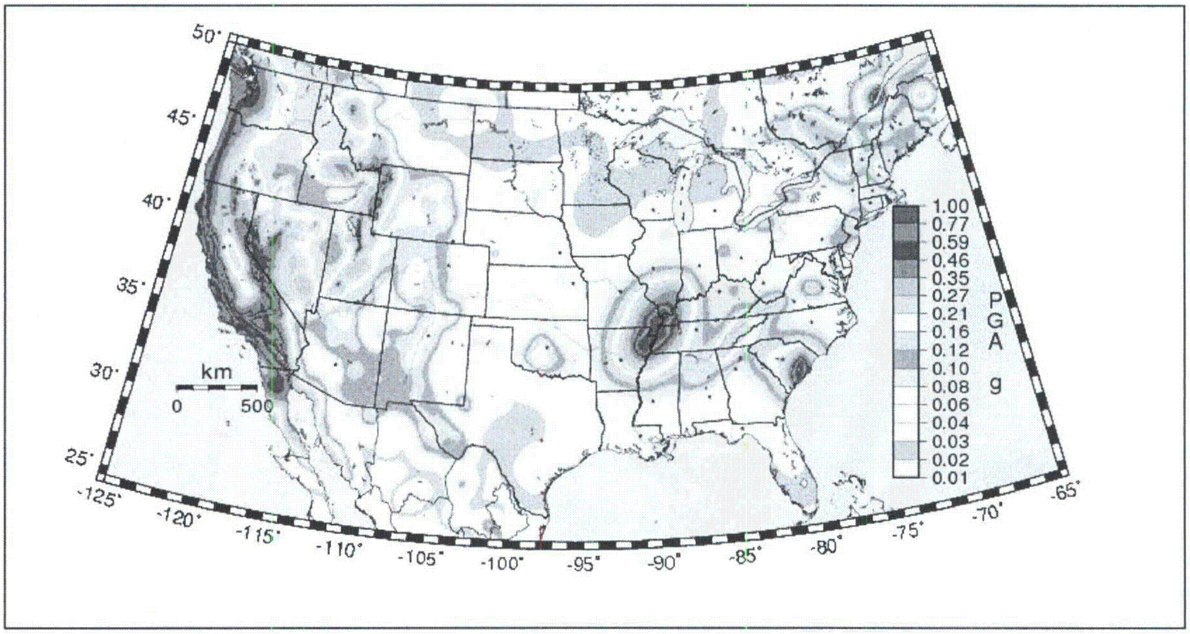


Figure 2. Peak Horizontal Acceleration (%*g*) for 2-Percent Probability of Exceedance in 50 Years for Conterminous United States. Source: USGS.

Table 1. List Of Operating Nuclear Power Plants Located Within the Central And Eastern United States.

Plant	Docket Number	Plant	Docket Number
Arkansas Nuclear 1	05000313	Millstone 2	05000336
Arkansas Nuclear 2	05000368	Millstone 3	05000423
Beaver Valley 1	05000334	Monticello	05000263
Beaver Valley 2	05000412	Nine Mile Point 1	05000220
Braidwood 1	05000456	Nine Mile Point 2	05000410
Braidwood 2	05000457	North Anna 1	05000338
Browns Ferry 1	05000259	North Anna 2	05000339
Browns Ferry 2	05000260	Oconee 1	05000269
Browns Ferry 3	05000296	Oconee 2	05000270
Brunswick 1	05000325	Oconee 3	05000287
Brunswick 2	05000324	Oyster Creek	05000219
Byron 1	05000454	Palisades	05000255
Byron 2	05000455	Peach Bottom 2	05000277
Callaway	05000483	Peach Bottom 3	05000278
Calvert Cliffs 1	05000317	Perry 1	05000440
Calvert Cliffs 2	05000318	Pilgrim 1	05000293
Catawba 1	05000413	Point Beach 1	05000266
Catawba 2	05000414	Point Beach 2	05000301

Table 1. List Of Operating Nuclear Power Plants Located Within the Central And Eastern United States.			
Plant	Docket Number	Plant	Docket Number
Clinton	05000461	Prairie Island 1	05000282
Comanche Peak 1	05000445	Prairie Island 2	05000306
Comanche Peak 2	05000446	Quad Cities 1	05000254
Cooper	05000298	Quad Cities 2	05000265
Crystal River 3	05000302	River Bend 1	05000458
D.C. Cook 1	05000315	Robinson 2	05000261
D.C. Cook 2	05000316	Saint Lucie 1	05000335
Davis-Besse	05000346	Saint Lucie 2	05000389
Dresden 2	05000237	Salem 1	05000272
Dresden 3	05000249	Salem 2	05000311
Duane Arnold	05000331	Seabrook 1	05000443
Farley 1	05000348	Sequoyah 1	05000327
Farley 2	05000364	Sequoyah 2	05000328
Fermi 2	05000341	South Texas 1	05000498
FitzPatrick	05000333	South Texas 2	05000499
Fort Calhoun	05000285	Summer	05000395
Ginna	05000244	Surry 1	05000280
Grand Gulf 1	05000416	Surry 2	05000281
Harris 1	05000400	Susquehanna 1	05000387
Hatch 1	05000321	Susquehanna 2	05000388
Hatch 2	05000366	Three Mile Island 1	05000289
Hope Creek 1	05000354	Turkey Point 3	05000250
Indian Point 2	05000247	Turkey Point 4	05000251
Indian Point 3	05000286	Vermont Yankee	05000271
Kewaunee	05000305	Vogtle 1	05000424
La Salle 1	05000373	Vogtle 2	05000425
La Salle 2	05000374	Waterford 3	05000382
Limerick 1	05000352	Watts Bar 1	05000390
Limerick 2	05000353	Wolf Creek 1	05000482
McGuire 1	05000369		
McGuire 2	05000370		

3.1 <u>Seismic Core-Damage Frequency Estimates</u>

Approximate SCDF estimates were developed by integrating the mean seismic hazard curve and the mean plant-level fragility curve for each NPP. This method, developed by Kennedy (1997), is discussed in Section 10.8.9 of AMSE/ANS RA-Sa-2009 and has previously been used by the staff in the resolution of GI-194, "Implications of Updated Probabilistic Seismic Hazard Estimates" and is the basis for the seismic performance-based approach for determining the site SSE and described in Regulatory Guide 1.208. Appendix A provides a detailed explanation of the method and its implementation.

The approach used in the Safety/Risk Assessment to approximate SCDF uses the available seismic risk information from IPEEEs and is computationally efficient. Computational efficiency is an important consideration because SCDFs need to be generated for each operating NPP located in the CEUS using various seismic hazard estimates to fully assess the implications of GI-199. However, the following are some recognized limitations to the approach:

- The integration of the mean seismic hazard curve and the mean plant-level fragility curve is not equal to the mean SCDF; accordingly, the SCDF estimates produced by the approach are point estimates. However, the numerical criteria in MD 6.4 are posed in terms of mean values.

- The approach does not provide a quantitative estimate of the parametric uncertainty in the SCDF. It should be noted that the mean seismic hazard curves produced by the USGS do not explicitly include uncertainty information.

- The approach does not provide any insight into which SSCs are important to seismic risk. This knowledge is needed if a regulatory analysis is required because it provides the basis for postulating plant backfits and conducting a value-impact analysis.

SCDF estimates were produced using three sets of mean seismic hazard curves that have been generated at various times and by various organizations as follows:

1. Electric Power Research Institute, 1989,
2. Lawrence Livermore National Laboratory, 1994.
3. NRC based on U.S. Geological Survey, 2008.

The following eight SCDF estimates were developed from each set of seismic hazard curves:

1. $SCDF_{pga}$ – integration of the pga-based seismic hazard and plant-level fragility curves.

2. $SCDF_{10}$ – integration of the 10-Hz seismic hazard and plant-level fragility curves.

3. $SCDF_5$ – integration of the 5-Hz seismic hazard and plant-level fragility curves.

4. $SCDF_1$ – integration of the 1-Hz seismic hazard and plant-level fragility curves.

5. $SCDF_{max}$ – maximum of the $SCDF_{pga}$, $SCDF_{10}$, $SCDF_5$, and $SCDF_1$ estimates.

6. $SCDF_{avg}$ – simple average of the $SCDF_{pga}$, $SCDF_{10}$, $SCDF_5$, and $SCDF_1$ estimates.

7. $SCDF_{IPEEE}$ – weighted average of the $SCDF_{pga}$, $SCDF_{10}$, $SCDF_5$, and $SCDF_1$ estimates, where the weights were obtained from Appendix A of NUREG-1407 ($SCDF_{pga}$ was weighted by one-seventh and the other SCDF estimates were weighted by two-sevenths).

8. $SCDF_{wl}$ – SCDF estimate based on the weakest link model described in Appendix A

3.2 Seismic Hazard Curves

As discussed earlier, the approach taken in the Safety/Risk Assessment was to assess changes in seismic hazard estimates with respect to previous estimates and then evaluate any risk significance of those changes using the Generic Issues decision framework. To proceed, it is necessary to develop both a current estimate of seismic hazard and an estimate of change in hazard for each NPP site of interest. This requires the specification of seismic hazards using current tools (i.e., the U.S. Geological Survey [USGS] hazard model results discussed below) and previous seismic hazard estimates that were considered to be acceptable. For this assessment, the seismic hazard estimates developed by EPRI-SOG (1989) and Lawrence Livermore National Laboratory (LLNL) (NUREG-1488, 1993) were used as the "baseline" cases from which changes could be evaluated. Both the EPRI and LLNL hazard results were identified as acceptable for use in the IPEEE evaluations, and the resulting SCDF values (either implied or explicitly computed) were deemed acceptable at the time. The results of the current SCDF and Delta-SCDF computations are discussed in more detail in subsequent sections.

The estimates of seismic hazard used in this Safety/Risk Assessment were obtained using the seismic hazard model developed by the USGS available during the fall of 2008. Other recent comprehensive seismic hazard studies have been conducted at various locations in the CEUS. Examples of these studies include the Trial Implementation Program (TIP) conducted by LLNL for NRC (NUREG-6607, 2002), a study for the South Carolina Department of Transportation (SCDOT), and a study performed for the Tennessee Valley Authority (TVA) Dam Safety Analysis Program. Unfortunately, these studies focused on small regions (or individual sites) and would not be useful for a systematic evaluation of all NPP sites in the CEUS.

As stated earlier in the Background section of this report, industry has updated the EPRI-SOG (1989) seismic source models as well as the ground motion prediction models for the CEUS in support of the ESP and Combined License (COL) applications submitted to NRC. These updated probabilistic seismic hazard analysis (PSHA) estimates would provide the staff with an ideal comparison to the earlier PSHA estimates; however, under program element 3A of the MOU, industry has only provided a very limited amount of information for NRC staff to use. As a result, the staff has primarily used the 2008 version of the USGS hazard model although it also evaluated the seismic hazard results provided by industry and submitted as part of the ESP and COL applications.

This USGS seismic hazard model has been developed and refined over a number of years (Frankel et al., 1996; 2002; Peterson et al., 2008). The USGS National Seismic Hazard Mapping Program follows a structured process to develop the seismic hazard models and computational programs used in the development of the national seismic hazard maps. This process involves a series of regional workshops used to elicit information and data from the research community and includes internal and external peer review of the resulting model. The results of this process are seismic hazard estimates for a dense grid of locations in the United States that are used as the basis for seismic design parameters in the current building codes (see Figure 2 for an example). Although not specifically designed to conform to the guidelines for performing high-level seismic hazard studies outlined in the Senior Seismic Hazard Analysis Committee (SSHAC) report (NUREG/CR-6372, 1997), the USGS process possesses many of the attributes of a Level 3 study as discussed in the SSHAC report. Likewise, although the EPRI-SOG (1989) study predated the SSHAC guidelines, the study had many of the attributes of a Level 4 study, and the updates being performed for the ESP and COL submittals are consistent with the SSHAC (Level 2) guidelines. The USGS seismic hazard models have not been used to site critical facilities such as NPPs although the NRC staff and industry have used the USGS hazard results for comparison to the EPRI-SOG models submitted in support of the ESP and COL applications. Recent regional or site-specific studies such as the TIP, SCDOT, and TVA studies mentioned above have been evaluated during the development of the USGS model as well as the updated EPRI-SOG model used in the ESP-COL applications.

For this assessment, the 2008 version of the USGS hazard model was used to compute seismic hazard estimates for individual plant locations (defined by latitude and longitude). For multiunit sites, the computation location was defined as the approximate center of the nuclear complex. The calculations assumed rock site conditions with near-surface shear wave velocity of 2,500 meters/second. Not all NPP sites can be reasonably represented as having hard rock site conditions. The definition of site type for individual units was generally consistent with the generic site classifications contained in the EPRI-SOG study (1989). For plants not evaluated in the EPRI-SOG (1989) study, the Final Safety Analysis Report (FSAR) was consulted to define a representative generic site classification. For any soil sites that had site-specific site amplifications available, those functions were used in lieu of the generic functions. Table B-2 in Appendix B summarizes the assumed site-type classifications for each NPP site.

As part of its preparation for submitting COL and ESP applications, industry has refined its site-specific amplification functions for many if not all of the CEUS NPP sites. However, the NRC staff has had access to only a few sites in addition to the ESP and COL sites that are collocated with a currently operating NPP. These site amplification functions can be quite different even for sites located very close together. Different assumptions regarding site amplification functions can have a very significant impact on hazard results (and subsequently on risk metrics). Figure B-5 in Appendix B illustrates this effect.

Seismic hazard estimates for each site were computed for four spectral frequencies (peak ground acceleration or PGA, 10, 5, and 1 Hz). Figure 3 illustrates representative results for rock hazard at the Ginna NPP site. Note that the hazard curves (H(*a*)) are monotonically decreasing and about linear in log-log-space. This figure illustrates the general (but not universal) characteristics of the comparison for many plants. Specifically, the latest USGS results are greater than the 1989 EPRI-SOG results but similar to, or in some cases less than, the 1993 LLNL results. Appendix B contains additional details on the computation of seismic hazard and additional comparisons.

As described in the next section, the results of the IPEEE program were utilized to develop fragility estimates to use with the seismic hazard results to produce plant-specific seismic CDF estimates. However, the IPEEE results represent the plant-level fragility in terms of PGA only; specifically, either directly or indirectly as a high confidence of low probability of failure PGA value ($HCLPF_{PGA}$). It is recognized that, at the plant level, the design response spectrum varies with frequency (Hz) (see Figure 4) and different elements within the plant may respond to different frequencies. As a result, it is desirable to estimate a frequency-dependent SCDF value over a range of frequencies of interest. This was accomplished by noting that as part of the IPEEE submittals, each NPP defined a review-level earthquake (RLE) spectral shape that was used in the review and analysis process. Table B-2 in Appendix B summarizes the IPEEE evaluation method, high confidence of low probability of failure (HCLPF) value, and RLE spectral shape for each NPP in the CEUS.

By anchoring the RLE spectrum to the $HCLPF_{PGA}$ –value and knowing the ratio between the spectral values of interest (10, 5, and 1 Hz) and PGA in the RLE spectrum, it is possible to compute $HCLPF_{10Hz}$, $HCLPF_{5Hz}$, and $HCLPF_{1Hz}$ values in addition to $HCLPF_{PGA}$. Figure 4 illustrates this procedure. Those plants that performed a seismic margins analysis (SMA) as part of the IPEEE evaluation generally utilized a smooth, broad-band RLE spectrum (NUREG-0098 or similar). However, for the plants that performed a seismic probabilistic risk assessment (SPRA), the RLE spectrum was generally based on a site-specific uniform hazard spectrum (UHS). In some cases this UHS fell below the plant-specific safe shutdown earthquake (SSE) (or design basis) spectrum at lower frequencies, implying the $HCLPF_{1Hz}$, for example, would be well below the design basis value. Figure 4 shows this effect. The NRC staff conducting this evaluation believes it is unlikely that the HCLPF would be ***less*** than the design value if the recommendations/requirements contained in the Standard Review Plan were followed. As a result, for this assessment, we have decided to test the spectral HCLPF values against the design values and have chosen the maximum of the two values (i.e., for each spectral frequency of interest: $HCLPF_{SA} = \max[RLE_{SA}, SSE_{SA}]$).

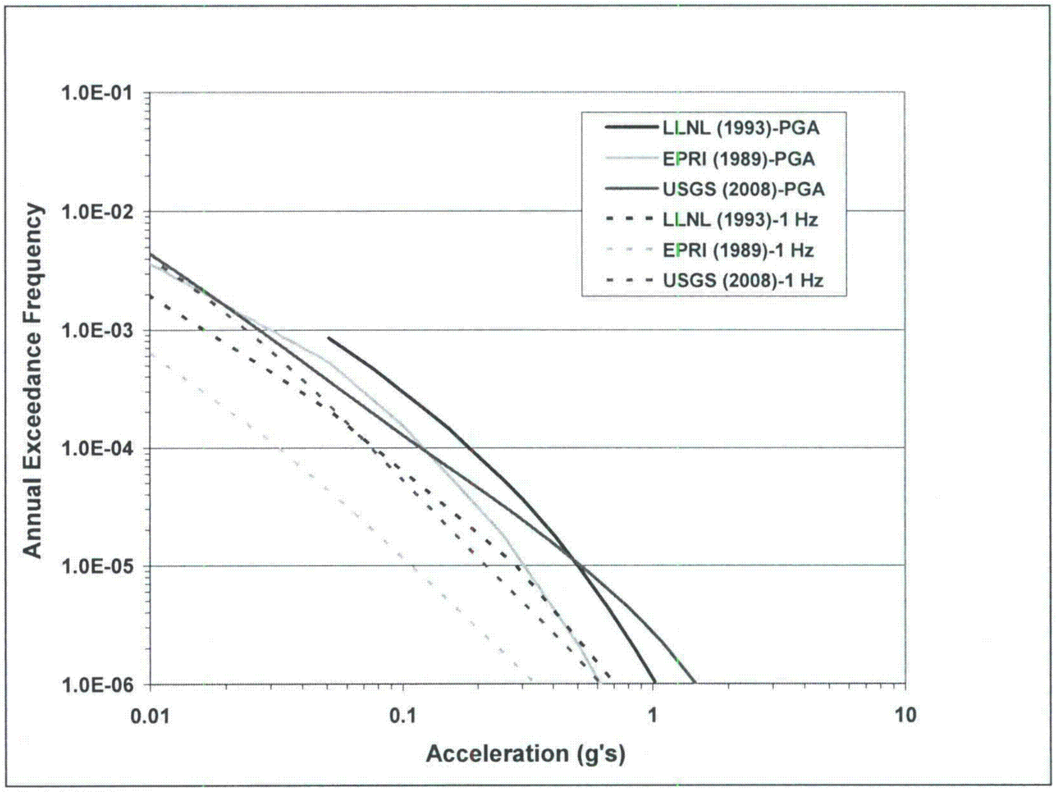


Figure 3. Comparison of Seismic Hazard Curves for the Ginna NPP Site. These curves were developed using the 2008 USGS seismic hazard model (red curves), the 1993 LLNL results (blue curves), and 1989 EPRI results (turquoise curves). Results for PGA are indicated by solid curves and for 1-Hz spectral acceleration by dashed curves.

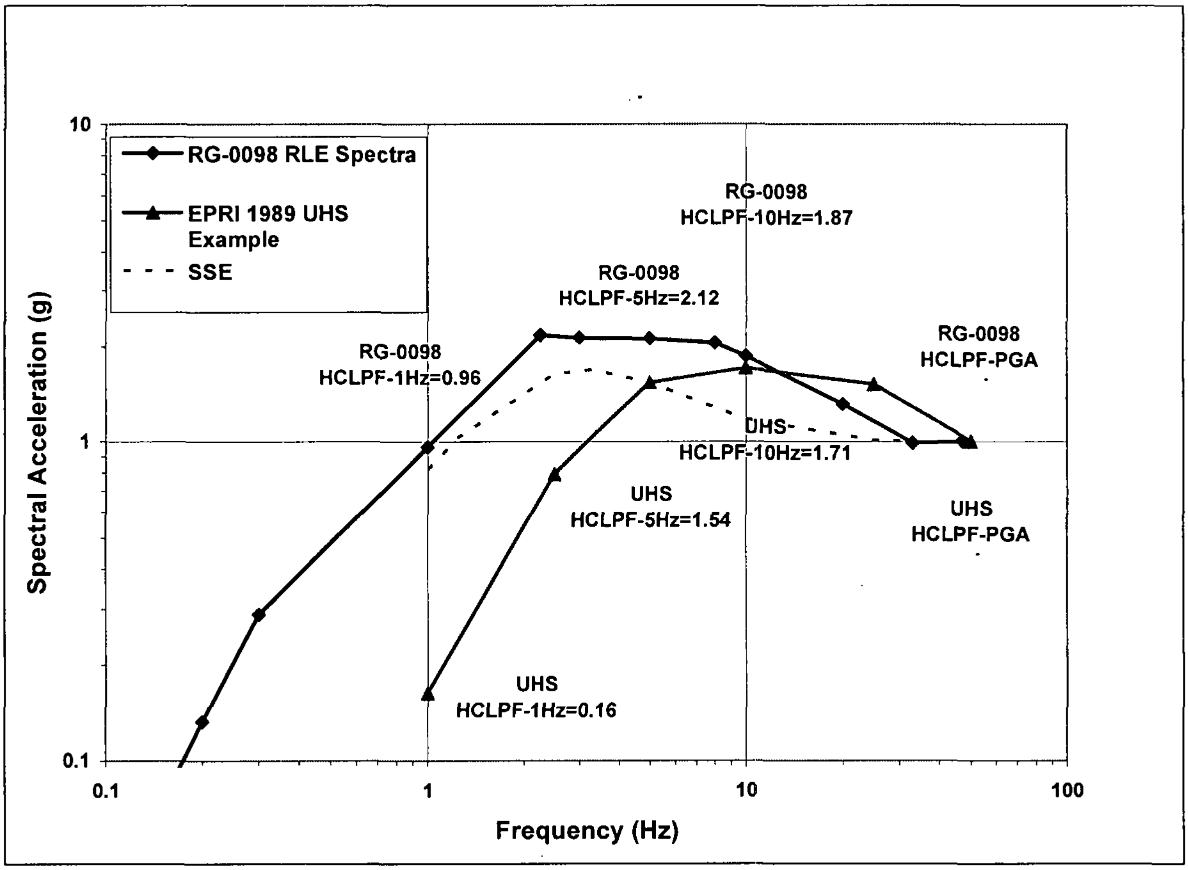


Figure 4. Illustration of Normalized Spectral Shapes Used in IPEEE Analyses. The blue curve is the spectral shape from RG-0098 that was used as the RLE in many IPEEE SMA assessments, the red curve is an example uniform hazard (UHS) spectrum similar to many used in the IPEEE SPRAs. The dashed curve is an example SSE spectrum (normalized). The HCLPF values for spectral frequencies other than PGA were assigned based on the ratio between the frequency of interest and PGA ($HCLPF_{10Hz}$ = 1.87*$HCLPF_{PGA}$ for the RG-0098 example shown here). For plants that used a UHS in the IPEEE assessment, the individual spectral HCLPF values (e.g., $HCLPF_{5Hz}$) were tested to see if they fell below the SSE spectrum at that frequency (SSE_{5Hz}); if so, the maximum of the two values was assigned.

3.3 Plant-Level Fragility Curves

The plant-level fragility curves were developed from information provided in the IPEEE submittals. It is recognized that plants may have made modifications that changed the plant-level fragility subsequent to completion of their IPEEEs; however, no regulatory requirement exists for plants to reflect the impact of such modifications in their IPEEEs (or, in fact, for plants licensed under 10 CFR Part 50 to maintain a PRA).

About one-third of the plants performed a SPRA as part of their IPEEE program. Licensees were not required to provide the actual SPRAs to NRC. Of the plants that performed SPRAs, about two-thirds provided plant-level fragility information (either in tabular or graphical format) in their IPEEE submittals. The remaining one-third of the SPRA plants provided SCDF estimates based on a variety of seismic hazard curves (EPRI 1989, LLNL 1994, or site-specific curves developed specifically for the IPEEE program). For these remaining plants, plant-level fragility values were back-calculated by matching the reported SCDFs and using engineering judgment. In cases where reasonable engineering judgments could not readily be made (e.g., the shape of the review-level ground motion ground spectrum), sensitivity studies were performed.

About two-thirds of the plants conducted a SMA as part of their IPEEE program. The figure of merit for an SMA is the plant-level HCLPF. Two SMA methodologies were recognized in Generic Letter 88-20, Supplement 4, for conducting an SMA—the EPRI methodology and the NRC methodology. Both methods utilize an RLE, which is specified in NUREG-1407 for each plant (listed in Table B-2 of Appendix B). In the EPRI methodology, two success paths are identified, where a success path consists of a selected group of safety functions capable of bringing the plant to a safe state after an earthquake larger than design basis and maintaining it there for 72 hours. The individual SSCs needed to accomplish each of the two success paths are then screened with respect to the RLE (if an SSC has a HCLPF that is less than the RLE, then the SMA uses the actual HCLPF; otherwise, the RLE is used). The individual SSC HCLPF values are then propagated through the success paths using simplified bounding logic to determine the plant-level HCLPF. The NRC approach uses fault tree logic (as opposed to success paths).

It is important to recognize that the actual plant-level HCLPF may not be determined by an SMA; that is, the RLE may be a lower bound for the actual plant-level HCLPF. This was acceptable for the IPEEE program because it was focused on identifying vulnerabilities and risk insights. However, it poses a challenge for the Safety/Risk Assessment because SCDF estimates based on the RLE may be conservative. This conservatism is opposed, however, by limitations in the basic SMA approach, which only treats random equipment failures (nonseismic failures) and operator errors in a simplified fashion.

Appendix C provides a detailed discussion of the development of the plant-level fragility curves and tabulates the fragility parameters used in the Safety/Risk Assessment.

4. RESULTS

4.1 SCDF Estimates

Using the 2008 USGS seismic hazard curves, all operating plants in the CEUS have SCDF less than or equal to 10^{-4} per year. This result confirms NRR's conclusion that currently operating plants are adequately protected against the change in seismic hazard estimates because the guidelines in NRR Office Instruction LIC-504, "Integrated Risk-Informed Decision Making Process for Emergent Issues," are not exceeded.

Generic Issues Program guidance contains numerical screening criteria in the form of an x-y plot, where the x-axis is the total baseline core-damage frequency and the y-axis is the change in core-damage frequency associated with the generic issue. The staff does not have estimates of the total core-damage frequency for each plant located within the CEUS (no information is available on external events such as fires, external floods, etc.). Moreover, establishing the baseline SCDF is problematic because this depends on which set of seismic hazard curves are used. Possible candidates include the 1989 EPRI-SOG results and the 1994 LLNL results because both were accepted by the staff for use in the IPEEE process. It must be noted that the licensing basis for plants located in the CEUS is based on deterministic analysis for design basis loads from the maximum earthquake level that is determined from historical data. Consequently, the licensing basis for these plants does not include a probabilistic assessment of seismic hazards or their potential impacts on plant risk. Figure 5 provides a comparison against the MD 6.4 criteria using both the EPRI data and the LLNL data to establish the baseline seismic risk. Continued evaluation is warranted for plants that lie in the shaded region of Figure 5.

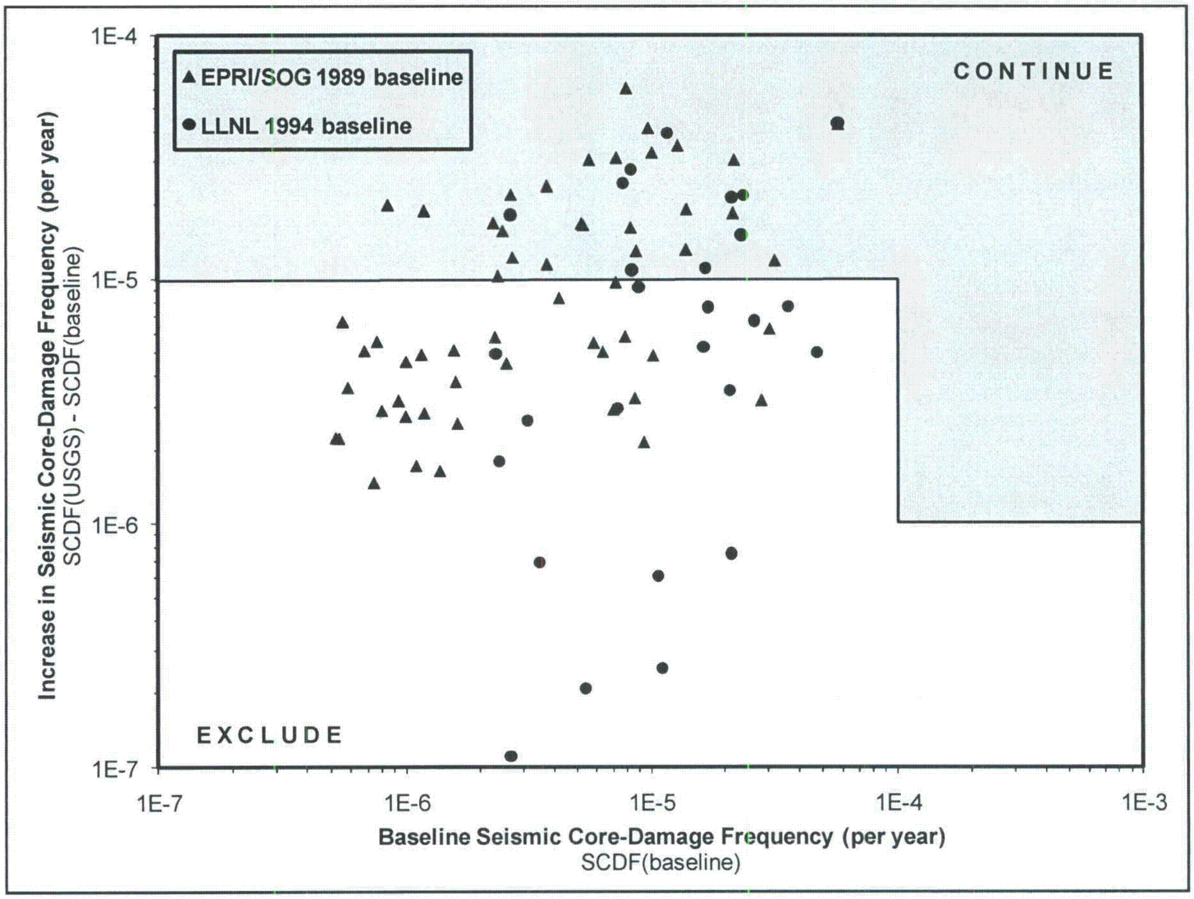


Figure 5. Comparison of Results from the Safety/Risk Assessment for GI-199 to the Screening Criteria in MD 6.4.

If the 1989 EPRI-SOG data are used to establish the baseline SCDF, then 36 plants lie in the "continue" region; if the 1994 LLNL data is used, only 11 plants lie in the "continue" region." These results do not change if the contribution from internal events, as computed by the staff's Standardized Plant Analysis of Risk (SPAR) models, is added to the baseline SCDF.

Another approach to review the results of the Safety/Risk Assessment is to develop fleetwide population variability distributions of the SCDF estimates. Figure 6 provides "box-and-whisker" plots of these distributions.

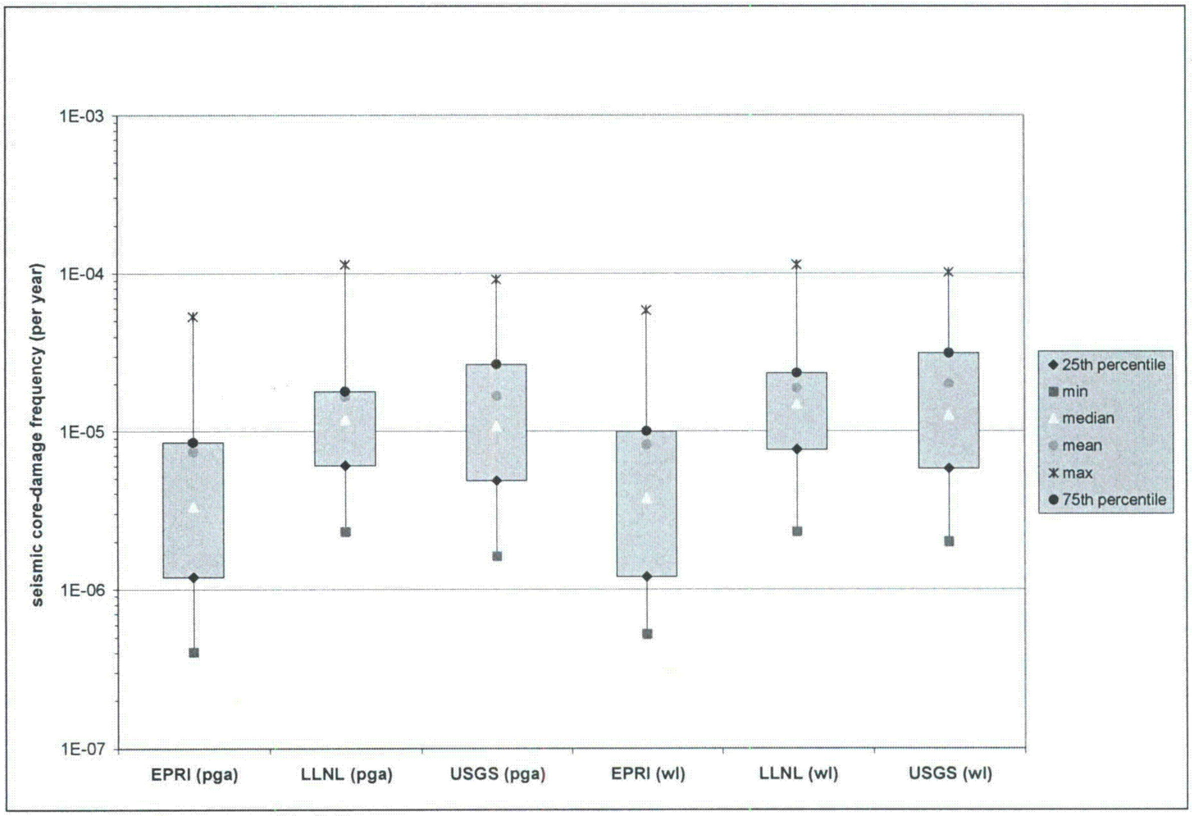


Figure 6. Fleetwide Population Variability Distributions of the Seismic Core-Damage Frequency Estimates.

Figure 6 indicates that the distribution of SCDF based on the 2008 USGS data is about the same as the distribution of SCDF based on the 1994 LLNL data. These results suggest that no change has been made in the fleetwide seismic risk since completion of the IPEEE program. However, Figure 6 must be carefully interpreted because the SCDF estimates at individual plants may have either increased or decreased. Figure 7 illustrates this observation by providing "box-and-whisker" plots of the distribution of the change in SCDF.

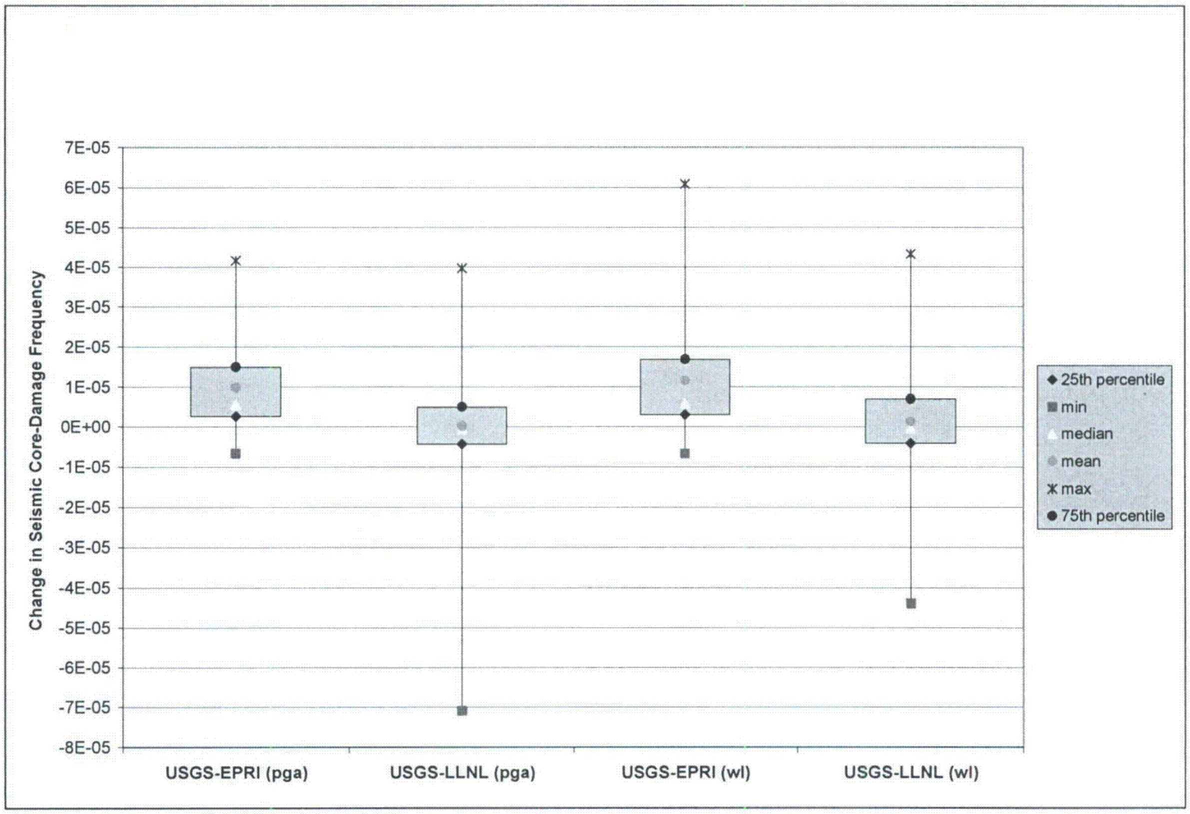


Figure 7. Fleetwide Population Variability Distributions of the Change in Seismic Core-Damage Frequency.

As a further aid to interpreting the results of the Safety/Risk Assessment, Figure 8 provides a plot that was constructed to simultaneously show the change in SCDF with respect to the 1989 EPRI data and the change in SCDF with respect to the 1994 LLNL data (this plot is termed the "delta-delta plot"). A "continue zone" was developed to identify plants where one change in SCDF is above 10^{-5} per year and the other change in SCDF is positive. Plants that lie in the "continue zone" are of potential interest because the SCDF based on the 2008 USGS seismic hazard data is greater than either of SCDF estimates based on the 1989 EPRI and 1994 LLNL seismic hazard data. The results of the Safety/Risk Assessment indicate that 24 plants lie in the "continue zone."

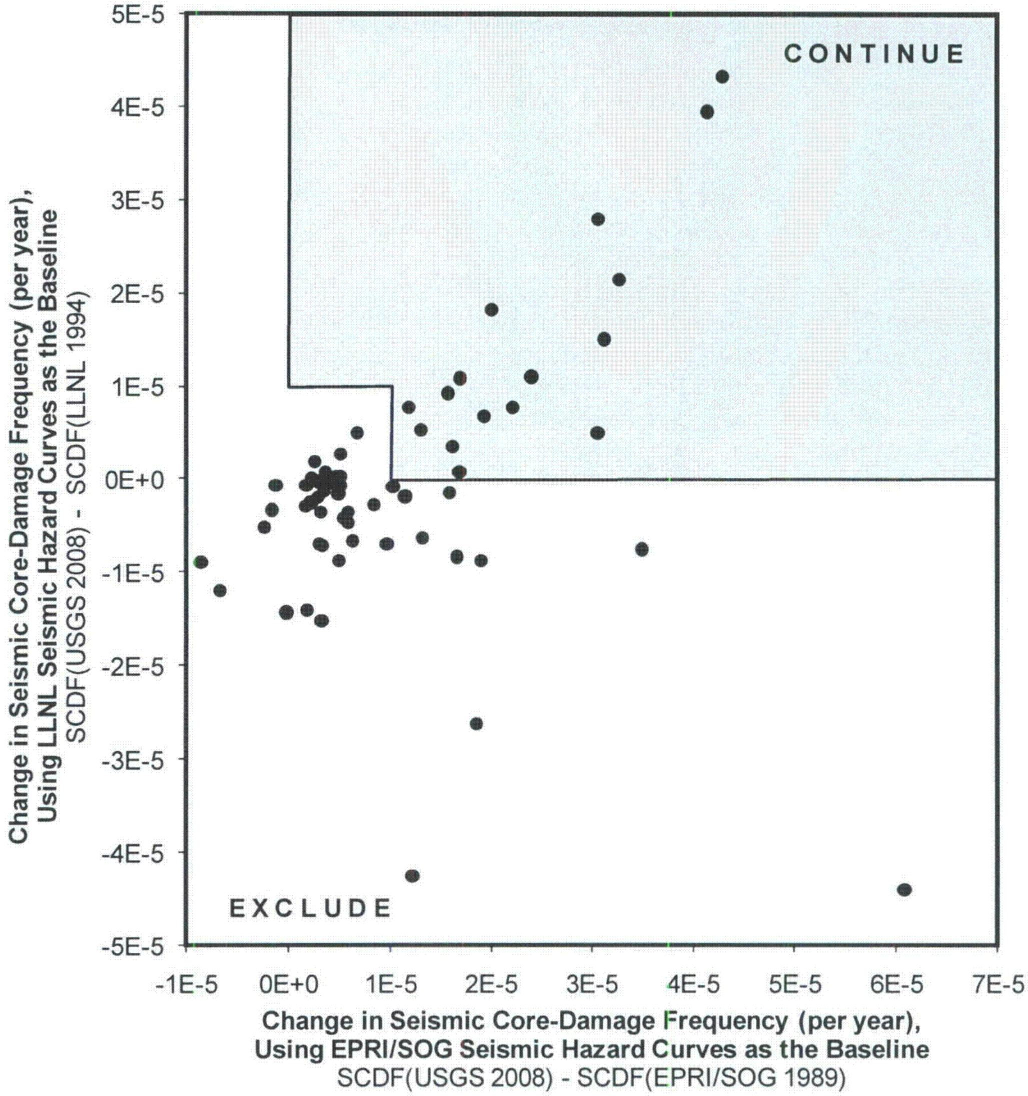


Figure 8. Change in SCDF with Respect to the 1989 EPRI and 1994 LLNL Seismic Hazard Data Sets Based on 2008 USGS Seismic Hazard Data (Delta-Delta Plot).

4.2 Evaluation of Changes in Seismic Hazard Estimates

To develop insights that may help in the Safety and Risk Assessment Stage, additional comparisons of the changes in seismic hazard were made. This evaluation of the potential significance of changes in seismic hazards was performed in a stepwise fashion posing a series of questions that, if answered in the negative, indicated no substantive change in the estimate of seismic hazard at a particular NPP. If the answer to the question was affirmative, the NPP was included in the next step of the assessment process.

Question 1. Does current staff guidance produce different design spectrum than the SSE?

The original development of seismic design bases for the existing reactor fleet was deterministic and not consistent with current practice. This does not necessarily mean that the seismic design basis (the Safe Shutdown Earthquake, or SSE, spectrum) was, or is, deficient in some fashion. If the process currrently defined in Regulatory Guide 1.208 is applied to develop a seismic design basis spectrum (the Ground Motion Response Spectrum [GMRS]), will it be different than the SSE for an individual site? To try and answer this question, the GMRS developed using the USGS-based hazard estimates is compared to the SSE.

Note: RG 1.208 provides an alternative for use in satisfying the requirements set forth in 10 CFR 100.23. Specifically, RG 1.208 was developed to provide general guidance on methods acceptable to the NRC staff for (1) conducting geological, geophysical, seismological, and geotechnical investigations; (2) identifying and characterizing seismic sources; (3) conducting a probabilistic seismic hazard assessment (PSHA); (4) determining seismic wave transmission (soil amplification) characteristics of soil and rock sites; and (5) determining a site-specific, performance-based GMRS. RG 1.208 states that a PSHA in the CEUS must account for credible alternative seismic source models through the use of a decision tree with appropriate weighting factors that are based on the most up-to-date information and relative confidence in alternative characterizations for each seismic source. It recognizes that the seismic sources identified and characterized by the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute Seismic Owners Group (EPRI/SOG) have been used for studies in the CEUS in the past, and that the United States Geological Survey also maintains a large database of seismic sources for both the CEUS and the WUS which may be beneficial in identifying the seismic sources that are relevant to a given nuclear power plant site. Although the LLNL, EPRI/SOG, or the USGS seismic hazard curves used in the GI-199 Safety/Risk Assessment do not, as-is, meet the guidance in RG 1.208, they are adequate for determining if GI-199 should proceed to the Regulatory Analysis Stage of the GIP.

To perform this comparison, several assumptions are required. First, the site characteristics and amplification functions are assumed to be similar to those defined in EPRI NP-6935 and free-surface motions are developed. Second, the seismic spectrum can be characterized by two intervals—peak ground acceleration (PGA) and spectral acceleration averaged between 5 and 10 Hz ($SA_{Avg5\text{-}10}$). PGA has been widely used to develop fragility estimates and represents the performance of SSCs that are sensitive to inertial effects. For SSCs that are sensitive to in-structure response, the $SA_{Avg5\text{-}10}$ captures the loading characteristics. Third, the GMRS is computed as:

$$GMRS_{USGS}=SA_{USGS}(\text{at Annual Exceedance Frequency of } 10^{-4})*DF$$

where DF is a design factor given by:

$$DF=\max(1.0,\ 0.6*Ar^{0.8}) \text{ and}$$

Ar is the ratio between SA at 10^{-5} and 10^{-4} annual exceedance frequencies.

The screening was based on identifying those plants where $(GMRS_{USGS}/SSE)^{PGA} > 1$ **and** $(GMRS_{USGS}/SSE)^{SAAvg5\text{-}10} > 1$. Figure 9 shows the results—61 of the 94 plants in the CEUS study area have increased PGA and $SA_{Avg5\text{-}10}$ relative to the SSE. For a plant to be "screened-in" using this criteria, both points plotted in Figure 9 must lie above the GMRS/SSE =1 line for that plant. The same logic holds for Figure 10 as well.

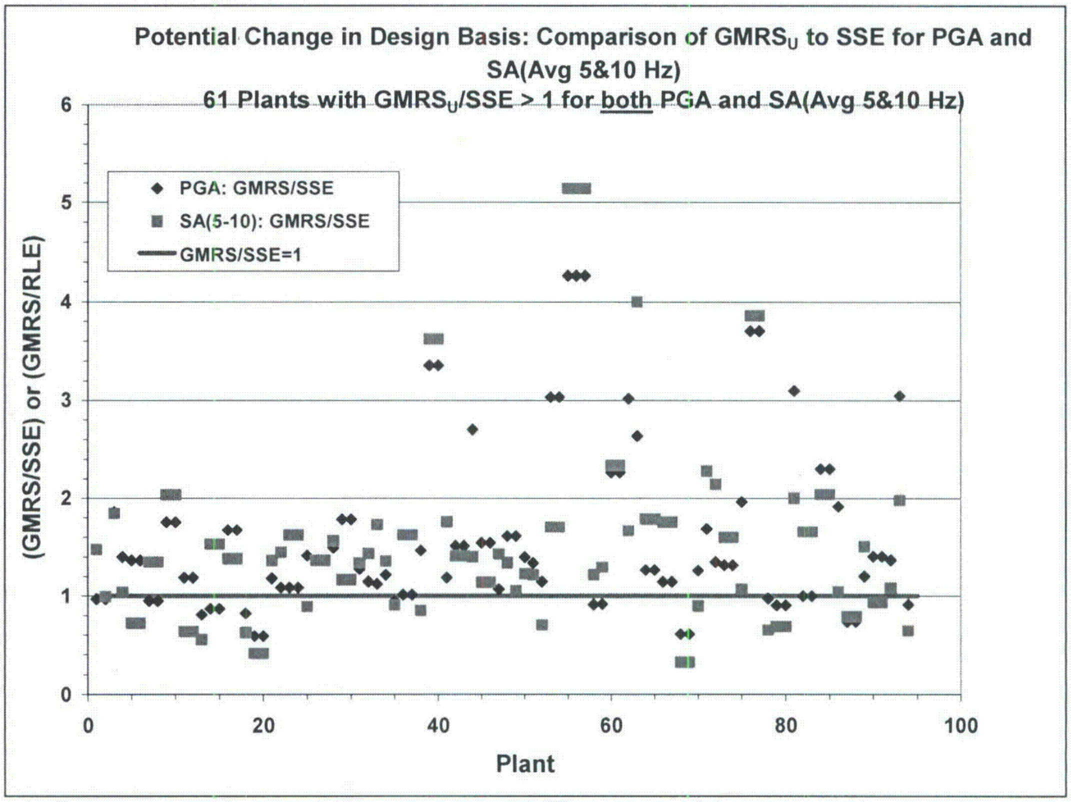


Figure 9. Comparison of GMRS$_{USGS}$ to SSE for PGA and SA$^{Avg5-10}$ for Plant Sites in CEUS. In this screening, 94 plants were evaluated and plotted.

Question 2. Does the current estimate of GMRS exceed the Review Level Earthquake (RLE) used in the IPEEE program?

All of the plants were evaluated under the IPEEE program, and many of them were evaluated for beyond-design basis earthquake loadings. The same strategy was employed as with the GMRS$_{USGS}$/SSE comparison. The SA$_{Avg5-10}$ values for the RLE were developed using the spectral ratios consistent with the spectral shapes suggested by NUREG-1742 (2001). Plants were identified that met the GMRS$_{USGS}$/SSE>1 criteria (Question 1) and where (GMRS$_{USGS}$/RLE)^{PGA} >1 **and** (GMRS$_{USGS}$/RLE)$^{SAAvg5-10}$ >1. Figure 10 shows the results—33 plants satisfy both the GMRS$_{USGS}$>SSE and GMRS$_{USGS}$>RLE screening criteria.

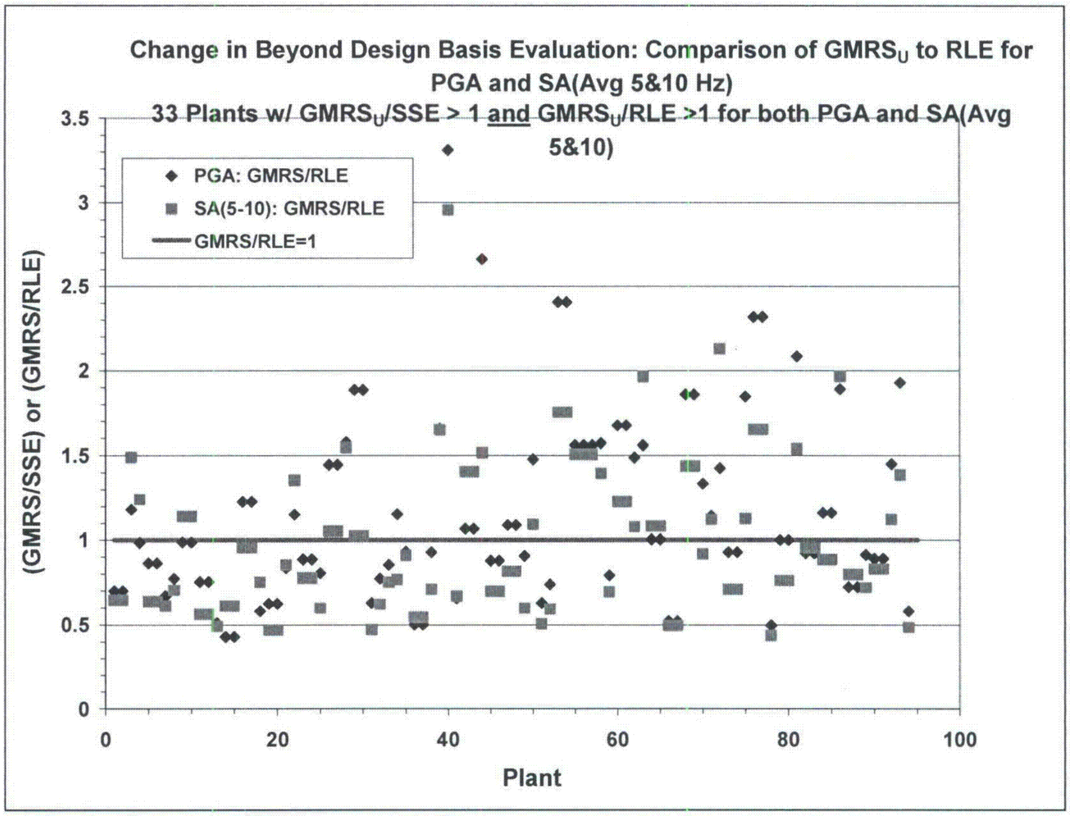


Figure 10. Comparison of $GMRS_{USGS}$ to RLE for PGA and $SA^{Avg5\text{-}10}$ for Plant Sites in CEUS.

Question 3. For those plants with increases in GMRS relative to the SSE and RLE is the change significant relative to previous seismic hazard estimates?

In addition to the SSE and RLE, previous seismic hazard estimates were developed as part of the LLNL and EPRI-SOG studies. It is appropriate to test the 2008 results against these previous estimates; if the latest hazard estimates fall within the range implied by the earlier studies, it seems reasonable to conclude no significant change has occurred. Conversely, if the latest estimates exceed both the LLNL and EPRI results, then a significant increase is likely in the hazard estimate. The same strategy was employed as with the $GMRS_{USGS}$/SSE and $GMRS_{USGS}$/RLE comparisons. Plants were identified that met the $GMRS_{USGS}$/SSE>1 criteria (Question 1), the $GMRS_{USGS}$/RLE >1 criteria (Question 2), and where $(GMRS_{USGS}/GMRS_{EPRI})^{PGA}$ >1, $(GMRS_{USGS}/GMRS_{EPRI})^{SAAvg5\text{-}10}$ >1 **and** $(GMRS_{USGS}/GMRS_{LLNL})^{PGA}$ >1, $(GMRS_{USGS}/GMRS_{LLNL})^{SAAvg5\text{-}10}$ >1. Figure 11 shows the results for $GMRS_{USGS}$>SSE, $GMRS_{USGS}$>RLE, and $GMRS_{USGS}$>$GMRS_{EPRI/LLNL}$ screening criteria. For a plant to be "screened-in" using this criteria, all four points plotted in Figure 11 must lie above the $GMRS_{USGS}/GMRS_{EPRI/LLNL}$=1 line for that plant. Of the 94 plants evaluated, 22 satisfy this screening criteria.

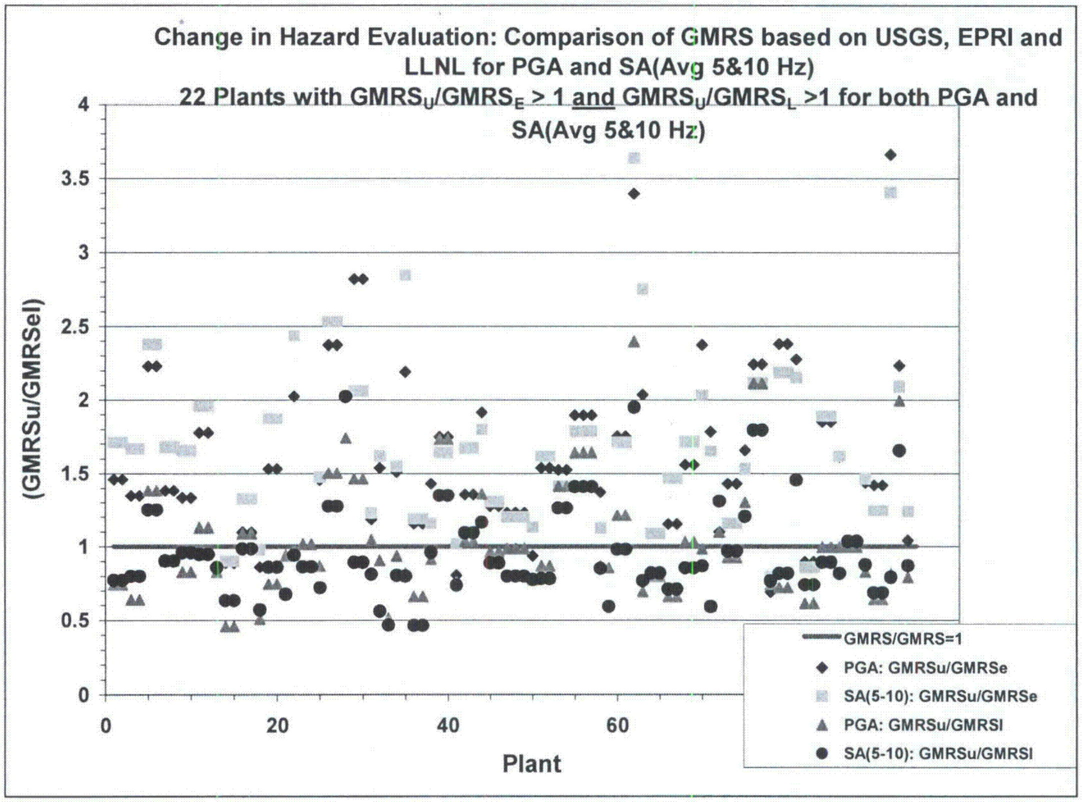


Figure 11. Comparison of $GMRS_{USGS}$ to $GMRS_{EPRI/LLNL}$ for PGA and $SA^{Avg5\text{-}10}$ for Plant Sites in CEUS.

Question 4. For those plants with increases in seismic hazard estimate, is there any significant change in risk metric?

To perform this assessment, the point estimates of mean seismic core damage frequency (SCDF) and change in SCDF (Δ-SCDF) are used. Sections 3 and 4 describe the development of these estimates. Use of SCDF and Δ-SCDF is consistent with MD 6.4 and will yield a general ranking of plants by risk. It must be recognized that the estimates are based on the available IPEEE data that are of variable quality and fidelity.

To compute an estimate of Δ-SCDF, a baseline SCDF must be defined. This is complicated because two sets of hazard curves exist that could be used for this computation (LLNL or EPRI-SOG). To try and alleviate this potential ambiguity, the delta-delta plot shown in Figure 8 was used. To answer Question 4, it is necessary to determine if any of the 22 plants identified in Question 3 appear in the "continue zone" of Figure 8. Of the 22 plants identified as having ground motion response spectrum (GMRS) values that exceed the SSE, exceed the RLE used in the IPEEE program, and exceed GMRS values based on previous EPRI and LLNL data, 21 appear in the continue zone of Figure 8.

5. DISCUSSION AND CONCLUSIONS

5.1 Discussion

The preceding sections summarize the analyses conducted for the Safety/Risk Assessment phase of the Generic Issues Process. It has been necessary to make a number of assumptions to perform these analyses. Prior to developing any conclusions, it is appropriate to specifically state some of the assumptions and limitations in the analyses as they impact some of the major conclusions.

- The use of the USGS-2008 seismic hazard model provides a representative estimate of the seismic hazard at specific NPP sites in the CEUS. However, this model has been developed and used for purposes other than critical facilities such as NPPs. The relative impact (and appropriateness) of certain assumptions within that model for the small annual AEF important for the safety evaluation of NPPs is still an open question. A different set of plants could be identified if a different hazard model was utilized.

- Very simplified, generic site response functions were assumed for the nonrock sites. This may produce very different estimates of seismic hazard (and consequently SCDF) relative to more accurate site specific response functions. At least some fraction of the sites identified in EPRI-SOG (1989) as "rock" are probably not appropriately classified as such when considering the most recent ground motion prediction equations.

- The Safety/Risk Assessment phase of GI-199 used a simplified approach based on combining plant-level fragility information developed from the IPEEE results with seismic hazard information to develop a point-estimate of SCDF. • The approach used to estimate SCDF does not provide any insight into which SSCs are important to seismic risk.

- The IPEEE studies were conducted to identify seismic vulnerabilities in the existing NPPs. In the GI-199 Safety/Risk Assessment, NRC staff is attempting to use that information for a different purpose—specifically to develop quantitative risk information. Significant differences in applicable information exist within the IPEEE results due to the different types of analyses conducted (PRA vs. full-, focused-, or reduced-scope SMAs) and screening level. •For a number of the plants that performed reduced-scope SMA analyses as part of the IPEEE program, little useful information exists regarding plant capacity.

- For many of the plants that performed a PRA and used a uniform hazard spectrum as the RLE-spectrum, NRC staff assumed that the HCLPF-point was at least equal to the SSE value for all structural frequencies.

- The IPEEE submittals generally provided limited information regarding the seismic capability of containments.

5.2 Conclusions

- **Seismic hazard estimates have increased:** Updates to seismic data and models indicate that estimates of the seismic hazard, at some operating nuclear power plant sites in the Central and Eastern United States, have increased.

- **There is no immediate safety concern:** Plants have seismic margin and the results of the GI-199 Safety/Risk Assessment confirm that overall seismic risk estimates remain small. GI-199 is not an adequate protection issue.

- **Assessment of GI-199 should continue:** Using available seismic hazard and plant seismic fragility information, the Safety/Risk Assessment found that the increase in core-damage frequency for about one-fourth of the currently operating plants is large enough to warrant continued evaluation under the Generic Issues Program. This conclusion is corroborated by the finding that, for many currently operating CEUS plants, a GMRS developed using the technical approach currently endorsed by the NRC staff is not bounded by the SSE (licensing basis) and exceeds previous "beyond design basis" evaluations (IPEEE RLE).

- **Additional information is needed to complete the assessment of GI-199:** Section 5.1 broadly discusses what additional information is needed to complete the assessment of GI-199. Specific additional information needs are listed below:

    - New site-specific seismic hazard curves: The staff is aware that EPRI has prepared new site-specific seismic hazard curves for many currently operating CEUS plants. In addition, new seismic hazard estimates for the CEUS will become available in late 2010 or early 2011 (these are a product of a joint NRC, DOE, USGS, and EPRI project). The hazard curves should cover a range of appropriate structural frequencies (PGA to 0.5 Hz), and be in a tabular, digital form.

    - New frequency dependent, site-specific amplification functions: Amplification functions are used to translate seismic motions from hard rock conditions to appropriate surface conditions. These functions should be consistent with the recent seismic evaluations performed by EPRI using updated seismic hazard results (see previous item), and be in tabular, digital form.

    - Current plant-level fragility information: The staff recognizes that many plants have been modified since completion of their IPEEEs, and believes that the plant-level fragility information used to complete the assessment of GI-199 should reflect the best available information. Specific information needed includes the median seismic capacity (C_{50}), the composite logarithmic standard deviation (β_C), and spectral ratios (relative to PGA) for 1, 5, and 10 Hz (at a minimum).

    - Plant-specific significant contributors to seismic risk: In order to progress with the Regulatory Analysis Stage, a comprehensive list of candidate plant backfits must be identified for subsequent value-impact analysis. One way to develop such a list is to consider the significant contributors to seismic core-damage risk and the approach used to identify them. It is also important to identify significant contributors to containment seismic performance.

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EPRI-1012965, 2005b, Program on Technology Innovation: Use of CAV in Determining Effects of Small Magnitude Earthquakes on Seismic Hazard Analyses, Electric Power Research Institute, Palo Alto, CA.

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Frankel, A. D., and 10 others, 2002, Documentation for the 2002 Update of the National Seismic Hazard Maps, U.S. Geological Survey Open-File Report 02-420, Denver, CO.

Kennedy, R.P., 1997, "Overview of Methods for Seismic PRA and Margins Including Recent Innovations," Proceedings of the Organization for the Economic Cooperation and Development/Nuclear Energy Agency Workshop on Seismic Risk, August 10-12, Tokyo, Japan.

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USNRC, Memorandum to M. Mayfield from F. Eltawila, "Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," dated June 9, 2005 (ML051600272).

Apostolakis, George

From:	Gilles, Nanette
Sent:	Thursday, September 08, 2011 3:58 PM
To:	Apostolakis, George
Cc:	Sosa, Belkys
Subject:	FW: FYI - Question from GI-199 Briefings

Commissioner – FYI

Nan

Nanette V. Gilles
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-1180
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EDO

From: Bowman, Gregory
Sent: Thursday, September 08, 2011 7:41 AM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike
Subject: FYI - Question from GI-199 Briefings

NRR was asked a question in response to the GI-199/North Anna briefings held last week for the Commissioners. The question pertained to activity in the seismic area between 1994 and 2008 (the time gap between development of the last two seismic hazard curves provided in the presentation). NRR provided the following:

> In March 1997, NRC issued RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" which introduced the concept of "Probabilistic Seismic Hazard Analysis," or PSHA. However, this RG was strictly intended in anticipation of future licensing applications. It was never intended as a tool to backfit existing plants that were licensed to RG 1.60 and earlier ground motion estimates based on the Housner and New Mark Spectra that were approved by the staff for older vintage plants. Around 2005 time frame, when we received requests for early site permits from North Anna, Clinton, and Grand Gulf, industry proposed the use of PSHA along with a performance-based approach to define the site-specific earthquake ground motion. That ultimately led to the development of RG 1.208. As such, RG 1.165 was, to my knowledge, never used in any licensing action. USGS hazard estimates were going down steadily from 1998, 2004, and 2008. This may explain, to some extent, the reason we didn't seem to aggressively pursue results of USGS seismic hazard predictions.

I'm passing this along for information only. If you have any questions, please give me a call.

L 13

Sanfilippo, Nathan

From: Weber, Michael EDO
Sent: Friday, September 09, 2011 5:29 PM
To: Haney, Catherine; Dorman, Dan; Sheron, Brian; Uhle, Jennifer; Bowman, Gregory; Erlanger, Craig; Frazier, Alan
Cc: Erlanger, Craig; Frazier, Alan; Brock, Kathryn; Grobe, Jack; Boger, Bruce; Leeds, Eric; Giitter, Joseph; Howe, Allen; Evans, Michele; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Johnson, Michael; Holahan, Gary; Merzke, Daniel; Sanfilippo, Nathan; Hayden, Elizabeth; Chokshi, Nilesh; Wert, Leonard; Hiland, Patrick; Li, Yong; Manoly, Kamal; Wertz, Trent; Martin, Robert; Thomas, George; Taylor, Robert; Franke, Mark; Boyle, Patrick; McCoy, Gerald; Wilson, George; Benner, Eric; Ordaz, Vonna; Weaver, Doug
Subject: FYI - Revised One Pager on North Anna Earthquake Issue
Attachments: One Pager on North Anna Earthquake Issue updated 9-9-2011 word doc (3).docx

Meena requested that the version of the "one pager" attached should be used (instead of the one circulated at 1638). Any questions, please contact Meena.

Thanks

From: Khanna, Meena NRR
Sent: Friday, September 09, 2011 4:38 PM
To: Bowman, Gregory
Cc: Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary; Merzke, Daniel; Sanfilippo, Nathan; Hayden, Elizabeth; Chokshi, Nilesh; Wert, Leonard; Hiland, Patrick; Skeen, David; Li, Yong; Manoly, Kamal; Wertz, Trent; Martin, Robert; Thomas, George; Taylor, Robert; Franke, Mark; Boyle, Patrick; McCoy, Gerald; Wilson, George; Benner, Eric; Ordaz, Vonna; Weaver, Doug
Subject: One Pager on North Anna Earthquake Issue

Greg,

As requested, attached is an update to the One Pager on the North Anna Earthquake issue.

Thanks,

Meena Khanna, Branch Chief
Mechanical and Civil Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation
(301)415-2150
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L14

NRR

Summary of Earthquake Information for the North Anna NPP as of Septmber 9, 2011

North Anna Design

The North Anna Nuclear Power Plant (NANPP) has two Safe Shutdown Earthquake (SSE) ground motions, one for structures, systems, and components (SSCs) located on top of rock, which is anchored at 0.12 g, and the other is for SSCs located on top of soil, which is anchored at 0.18 g. The NANPP has two corresponding Operating Basis Earthquake (OBE) ground motion spectra, anchored at 0.09 g for soil and 0.06 g for rock.

The current best estimate of the Peak Ground Acceleration (PGA) for the NANPP site is 0.26 g, which contains uncertainty. This estimate indicates that the ground motion likely exceeded the SSE response spectra for NANPP Units 1 and 2 (0.12 g) over a considerable frequency range. The estimated ground motion from the earthquake was not a surprise based on the combined operating license application (COLA) ground motion response spectrum for NANPP Unit 3. This preliminary estimate appears to validate the NRC's current seismic hazard assessment approaches and the basis for GI-199 reviews.

The licensee has retrieved its seismic instrumentation recordings located at different elevation levels from within the plant and has processed the initial information. Preliminary results from the seismic instrumentation indicate exceedance above the final safety analysis report (FSAR) design basis earthquake (DBE) at various frequencies, depending on the building, measurement direction, and elevation. The data will be used to inform the staff regarding actions necessary for restart as well as long term design verification.

The licensee is performing plant walk downs in accordance with RG 1.167, "Restart of a Nuclear Power Plant Shutdown by a Seismic Event," which endorses EPRI's "Guidelines for Nuclear Plant Response to an Earthquake" with conditions. The staff will assess the licensee's evaluation of SSCs that are most sensitive to ground motion in that frequency band.

Timeline

- On August 23, 2011, North Anna Power Station declared an Alert due to significant seismic activity onsite from an earthquake which had a measured magnitude of 5.8.
- The licensee conducted the 1st general walkdown of the plant as required by the North Anna Power Station abnormal procedure for seismic event.
- The licensee conducted the 2nd walkdown after the magnitude 4.5 aftershock.
- Preliminary readings of the Seismic Response Spectrum Recorder (scratch plate) and the magnetic tapes identified that the Design Basis Earthquake had been exceeded at certain frequencies. On August 26, the licensee declared all safety-related SSCs of Units 1 and 2 inoperable and issued a 10 CFR 72 Notification.

Post Event Response

- An Augmented Inspection Team arrived on site on August 31, 2011.

- The initial determination by the licensee was that the U1 and U2 reactor trip signals were initiated by a reactor protection system negative neutron rate flux trips that occurred prior to the LOOP.

- Preliminary data provided by the licensee indicate that the DBEs at different elevation levels have been exceeded in both the horizontal and vertical directions at different frequencies (see attached sample figure).

- The licensee's inspections have not identified any safety related equipment which failed during the event except for the 2H EDG. When it became evident that the DBE had been exceeded, the licensee declared all the safety related equipment in both units inoperable and took action to place the units in cold shutdown.

- Preliminary data from the licensee's investigation about the 2H EDG failure indicates that a cooling water system gasket may have been installed incorrectly.

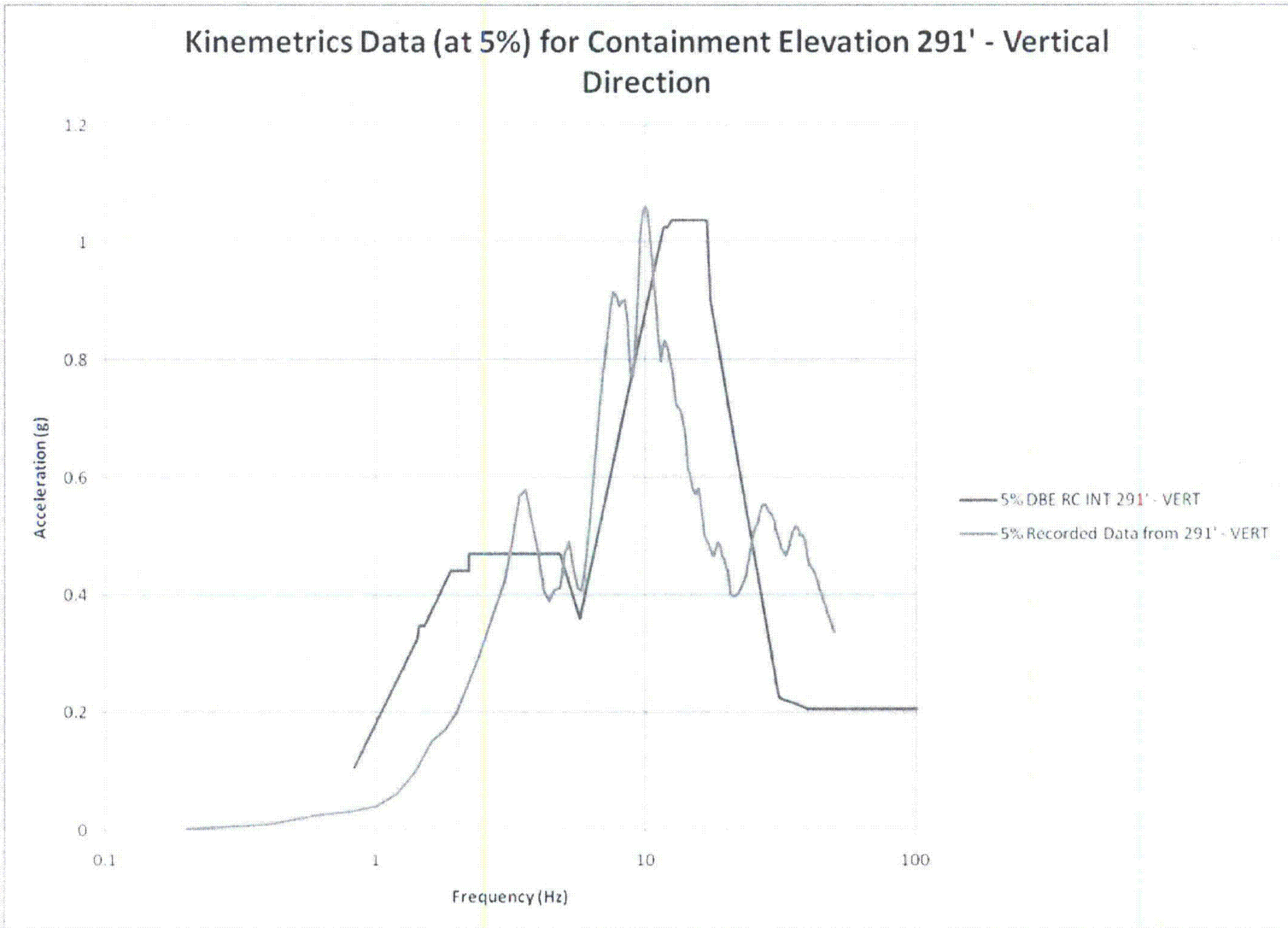
Path Forward

- A public meeting was held, at the licensee's request, on September 8, 2011. The licensee discussed their initial assessment of the earthquake's impact on the North Anna plant, and presented some of their future plans.

- Staff is developing an Action Plan that will contain the staff's expectations on topics or actions that will be reviewed prior to a plant restart decision

- Licensee will submit request for restart that will provide results of their inspections and readiness reviews.

- Appropriate regulatory vehicle to assure licensee actions are adequate is being considered.



The figure above, as submitted by the licensee, compares the observed ground motion in the vertical direction to the exceeded design motion at various frequencies. The blue curve is from the observed readings and the purple curve is the design spectrum at the same elevation level.

North Anna Independent Spent Fuel Storage Installation
Response to Earthquake

Background:

The North Anna Independent Spent Fuel Storage Installation (ISFSI) uses two spent fuel storage systems manufactured by Transnuclear (TN)

1) Twenty seven vertical TN-32 metal casks under a 10 CFR Part 72 site specific license. This system has a bolted closure lid with a pressure monitoring/alarm system, and stands freely on the ISFSI concrete pad. The design/licensing basis for the vertical TN-32 is controlled primarily by the North Anna ISFSI FSAR and NRC license (SNM-2507) and NRC certificate (1021). The FSAR defines the design acceleration values of 0.18g horizontal and 0.12g vertical, and sliding was not predicted to occur at these values.

2) Twenty six TN NUHOMS HD-32PTH horizontal storage modules (13 loaded) under a 10 CFR Part 72 general license. This system uses a welded-sealed canister and rests on horizontal rails inside the horizontal storage module. The design/licensing basis for the TN NUHOMS HD is controlled primarily by the separate TN-NUHOMS FSAR and NRC certificate (1030), as supplemented by additional site-specific evaluations that were performed by North Anna under 10 CFR 72.212. NUHOMS-HD components are designed to acceleration values of 0.3g horizontal and 0.2g vertical.

Event:

The North Anna ISFSI suffered minor damage from the earthquake:

1) Twenty five of the twenty seven TN-32 casks slid up to 4.5 inches on the concrete pad during the quake. Six cask sets (12 casks) were closer than the 16 foot separation distance specified in the FSAR. There was no damage to the pressure monitors in each cask and no pressure monitoring system alarms during or after the earthquake. There were no crack indications observed in the concrete pad or casks.

2) For the TN-NUHOMS modules, some slight damage was identified around the outlet vents and some surface cracking indications were noted. Additionally, some modules showed gaps between them of approximately 1.5" versus the required 1.0" maximum gap.

Preliminary Determination of Safety Significance:

The staff believes there is no immediate safety issue. The cask designs are robust and consider severe natural phenomena. As expected, the casks withstood the earthquake at North Anna. The spent fuel continues to be surrounded by several tons of steel and concrete, and sealed in an inert helium environment. Damage to concrete components appear to be cosmetic, and does not impact structural integrity or radiation shielding capability. Additionally, the fuel assemblies are designed to withstand a maximum of 4g axial load and 6g lateral load. Inlet and outlet vents were inspected and no exterior blockage was found. Radiation surveys indicate no changes to cask surface dose rates. Thermal performance measurements for all loaded casks found no abnormal temperature differences.

Additionally for the TN-32 casks, the requirement specifying a minimum distance of 16 feet between casks with a heat load greater than 27.1 kW was conservatively established so that the casks do not influence each other thermally and to allow for emplacement on the pad by the cask transporter. Currently, the two casks with the least separation (15 feet, 3.5 inches) are casks that had decay heats of 15.4 kW and 18.0 kW when loaded in 2000 and 2001, both well below the 27.1 kW requirement.

Licensee Response:

The licensee is following RG 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions" as a guide to perform their post-event assessment and has completed walkdowns of the ISFSIs.

The licensee reviewed this event for reportability under 10 CFR 72.75 (significant reduction in effectiveness of any spent fuel storage cask confinement system) and determined that the TN-32 displacement and NUHOMS-HD damage described above was not reportable.

The licensee contacted TN and provided them with all available pictures, data, and inspection results. TN requested that the licensee perform a more detailed inspection and evaluation of the current condition and sent a team to support this inspection.

NRC Response:

Item 10 of the AIT charter requires the AIT to "Assess the extent of any impact or damage to the Independent Spent Fuel Storage Installation from the seismic event." NMSS and Region II will continue to support the AIT and evaluate information related to the ISFSI to determine whether longer-term licensing or inspection actions are warranted for North Anna or generically.

On September 1, 2011, AIT completed a walk-down of the ISFSI Pads and has concluded that there are no indications of immediate safety issues associated with the movement of the vertical and horizontal ISFSI modules. Radiological conditions are normal and monitoring systems are functional. Damage as a result of the earthquake did not seem detrimental for the integrity of the casks.

On September 7, 2011, NMSS and Region II participated in technical discussions with the licensee to discuss near and long term ISFSI plans. NMSS will determine appropriate vehicle to ensure that licensee takes appropriate action.

Sosa, Belkys

From:	Sosa, Belkys
Sent:	Monday, September 12, 2011 2:51 PM
To:	Sanfilippo, Nathan
Subject:	RE: North Anna plan
Categories:	Red Category

thanks

EDO

From: Sanfilippo, Nathan
Sent: Monday, September 12, 2011 9:53 AM
To: Sosa, Belkys
Subject: North Anna plan
Importance: High

Belkys,

We expect North Anna to submit their plan in the next couple of weeks. At present, we don't have anything but the slides from the public meeting.

Thanks,
Nathan

L15

Astwood, Heather

From:	Castleman, Patrick
Sent:	Monday, September 19, 2011 3:48 PM
To:	Svinicki, Kristine
Cc:	Sharkey, Jeffry; Reddick, Darani; Astwood, Heather
Subject:	FW: FYI - Dominion Letter on North Anna
Attachments:	11-520 Earthquake Summary Report and Restart Plan Final.pdf

FYI

From: Bowman, Gregory
Sent: Monday, September 19, 2011 11:50 AM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike
Subject: FYI - Dominion Letter on North Anna

Dominion sent in the attached letter on Saturday. It provides their assessment of the impact of the earthquake on plant SSCs and the actions necessary to demonstrate the acceptability of the restart of North Anna Units 1 and 2.

I'm passing this along for information. The staff is in the process of reviewing the letter and hasn't made any conclusions on the technical content at this point.

Greg

L16

Apostolakis, George

From:	Gilles, Nanette
Sent:	Monday, September 19, 2011 11:57 AM
To:	Apostolakis, George
Cc:	Sosa, Belkys; Davis, Roger
Subject:	FW: FYI - Dominion Letter on North Anna
Attachments:	11-520 Earthquake Summary Report and Restart Plan Final.pdf

Commissioner – FYI – See below. I have saved this report on G – Seismic – North Anna.

Nan

Nanette V. Gilles
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-1180
Email: nanette.gilles@nrc.gov

From: Bowman, Gregory EDO
Sent: Monday, September 19, 2011 11:50 AM
To: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike
Subject: FYI - Dominion Letter on North Anna

Dominion sent in the attached letter on Saturday. It provides their assessment of the impact of the earthquake on plant SSCs and the actions necessary to demonstrate the acceptability of the restart of North Anna Units 1 and 2.

I'm passing this along for information. The staff is in the process of reviewing the letter and hasn't made any conclusions on the technical content at this point.

Greg

L17

ML112620151
public

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 17, 2011

10 CFR 100, Appendix A

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No.:	11-520
NL&OS/GDM	R3
Docket Nos.:	50-338/339
	72-16/72-56
License Nos.:	NPF-4/7
	SNM-2507

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
NORTH ANNA INDEPENDENT SPENT FUEL STORAGE INSTALLATION
SUMMARY REPORT OF AUGUST 23, 2011 EARTHQUAKE RESPONSE AND RESTART READINESS DETERMINATION PLAN

On August 23, 2011, at 1351 hours, with North Anna Power Station Units 1 and 2 operating at 100% power, a Magnitude 5.8 earthquake occurred approximately 5 miles from Mineral, Virginia. The epicenter was approximately 11 miles WSW of North Anna Power Station. Ground motion was felt and recognized as an earthquake by the Main Control Room operators at the station. The earthquake caused a series of reactor trip signals to both the Unit 1 and Unit 2 reactors, as well as a total loss of offsite power to the station. Per the Event Review Report, the "First Out" reactor trip signals for both units were "High Flux Rate Reactor Trip". Other than the trip signals and subsequent loss of offsite power, which were either directly or indirectly caused by the seismic event, safety systems in the plant responded as expected to the reactor trip and remained functionally undamaged and capable of performing their intended design functions. Separately, the 2H Emergency Diesel Generator developed a coolant leak and was subsequently manually secured. A Root Cause Evaluation of the leak is being performed. Based on evaluation of the US Geological Survey (USGS) data and plant-specific seismic data analysis, Dominion has confirmed that the August 23, 2011 earthquake exceeded the spectral and peak ground accelerations for the Operating Basis and Design Basis Earthquakes ("OBE" and "DBE", respectively) for North Anna Power Station Units 1 and 2.

Demonstration of North Anna Restart Readiness

Paragraph V(a)(2) of Appendix A to 10 CFR 100, states "if vibratory ground motion exceeding that of the Operating Basis Earthquake... Prior to resuming operations, the licensee will be required to demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public." Consistent with the regulatory requirement, Dominion met with the NRC staff on September 8, 2011 to: 1) provide an overview of the station response to the seismic event, 2) summarize the recorded data and analysis to date, and 3) discuss the station restart readiness assessment plan. As a follow-up to that meeting, we are providing detailed information to demonstrate that the post-earthquake analysis, inspection and testing activities that have been or will be completed by Dominion are sufficient to ensure that station structures, systems and components (SSCs) will continue to perform their required design functions such that restart of North Anna Units 1 and 2 may commence without undue risk to the health and safety of the public. Specifically, the enclosures to this letter collectively provide Dominion's North Anna Restart Readiness Assessment Plan as follows:

- Enclosure 1 Characterization of the North Anna Seismic Event of August 23, 2011
- Enclosure 2 Post-Earthquake Inspections of Plant Structures, Systems and Components
- Enclosure 3 Post-Earthquake Evaluation of the Reactor Vessel Internals
- Enclosure 4 Post-Earthquake Assessment of New and Irradiated Fuel
- Enclosure 5 Post-Earthquake Assessment of the Spent Fuel Storage Racks
- Enclosure 6 Post-Earthquake Evaluation of the Independent Spent Fuel Storage Installation (ISFSI)
- Enclosure 7 Post-Earthquake Impact Assessment on Engineering Programs
- Enclosure 8 Near-Term Actions to be Completed Prior to Unit Startup
- Enclosure 9 Long-Term Actions to be Completed After Unit Startup

Based on exceeding the station OBE and DBE seismic criteria, the North Anna Restart Readiness Assessment Plan was based on the guidance contained in the following documents:

1. RG 1.166, *Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions,* dated March 1997,
2. RG 1.167, *Restart of a Nuclear Power Plant Shut Down by a Seismic Event,* dated March 1997, *and*
3. EPRI NP-6695, *Guidelines for Nuclear Plant Response to an Earthquake,* dated December 1989.

EPRI NP-6695, which is endorsed by RG 1.167, bases post-event actions on the EPRI Damage Intensity Scale, which is dependent upon the level of damage observed, and the long-term actions on whether the equipment and structures were subjected to loads greater than the Safe Shutdown Earthquake (SSE) (i.e., DBE). If the level of damage to nuclear power plant equipment and structures observed during the post-shutdown inspections is found to be significant (i.e., corresponds to a Damage Intensity 3, which is based on the EPRI damage scale for nuclear plant facilities given in EPRI NP-6695 Tables 2-1 and 2-2), then the EPRI guidance would direct that long-term evaluations be performed and completed prior to restart. Completed plant inspections of North Anna's

SSCs performed in accordance with EPRI NP-6695 indicate an EPRI Damage Intensity of 0.

EPRI TR-100082, *Standardization of the Cumulative Absolute Velocity*, which is referenced in RG 1.166, provides criteria for calculating a cumulative absolute velocity (CAV) value which is an indicator of expected damage level from a particular earthquake spectrum. The threshold value for the onset of damage, provided in EPRI TR-100082 and mirrored in RG 1.166, is 0.16g-sec. Per EPRI TR-100082, "the adjusted CAV threshold is about a factor of five lower than the lowest CAV value associated with documented damage to an industrial/power facility. It is also about a factor of three lower than the lowest CAV value associated with documented damage to buildings of good design and construction."

In accordance with EPRI NP-6695, Appendix A, the criterion for determining if the OBE has been exceeded is independent of the plant's design OBE and SSE ground response spectra. For plants with a low SSE ground response spectrum (i.e., less than 0.2g,) it is possible to exceed the OBE and even the SSE ground response spectrum and not exceed the OBE CAV criterion if the damage parameters are less than the limit values (i.e., peak 5 percent damped ground motion spectral acceleration less than 0.2g or CAV less than 0.3 g-sec). The 0.3g-sec was later changed to 0.16 g-sec per EPRI TR-100082 using a different calculation methodology.

The CAV values calculated for the North Anna seismic event were below the very conservative threshold of 0.16g-sec defined in NRC RG 1.166 for OBE exceedance in the E-W and vertical directions, and the calculated CAV value exceeded the threshold by about 10% in the N-S direction. Because the plant-specific CAV values only slightly exceeded the specified OBE CAV limit in one of the three directions, no significant physical or functional damage would be expected to either safety related or non-safety related SSCs. This expectation is consistent with the findings of the comprehensive inspections performed on plant SSCs following the earthquake.

Specifically, completed post-shutdown plant walkdowns and inspections have not identified any significant physical or functional damage to safety-related plant SSCs and only limited damage to non-safety related, non-seismically designed SSCs (e.g., Generator Step-Up Transformer bushings). The lack of any significant physical or functional damage to safety-related SSCs and the limited damage to non-safety related systems are consistent with an EPRI Damage Intensity of 0, the indicator of least damage. Despite the lack of evidence of any physical or functional damage to safety-related plant SSCs, a conservative decision was made to perform comprehensive and methodical visual inspections of plant SSCs and to perform expanded inspections and tests in accordance with an EPRI Damage Intensity 1 versus the observed 0. Detailed discussion of the plant walkdowns and inspections is provided in the enclosures.

In addition to the post-earthquake plant walkdowns and inspections noted above, the following reviews and evaluations were also performed: Review of Recorded Vibratory

Data from Seismic Monitoring Instruments, Service Water Reservoir & Main Dam Review, ISFSI Pad and Cask Review, New Fuel and Spent Fuel Pool Inspection Review, Reactor Vessel Internals Evaluation, Design and Licensing Basis Review, Engineering Programs Review, Periodic Tests Review and Underground Piping Review.

Plant Seismic Design Considerations

As noted above, the response spectra developed from the time-histories of the recordings at the Reactor Containment basemat indicate that the Mineral, Virginia earthquake of August 23, 2011 exceeded the North Anna OBE and DBE ground response spectra that are currently specified as the licensing and design bases for North Anna Power Station Units 1 and 2. However, the CAV values and the effective strong motion durations were small.

From our overall assessment, the earthquake exceeded, on average, the North Anna DBE spectral accelerations in the more damaging 2 to 10 HZ frequency range by about 12% in one horizontal orientation, 21% in the vertical orientation, and not at all in the other horizontal direction. The North Anna DBE ground spectra for rock and soil founded structures are anchored to 0.12g and 0.18g peak ground accelerations (PGA) respectively in the horizontal direction. While the above exceedances are at the basemat of the containment, a comparison of the response spectra from the recorded time-histories of this earthquake to the calculated DBE in-structure response spectra (ISRS) at the operating deck of the containment structure (elevation 291') shows that the exceedances are smaller and only at certain narrow frequency bands in the 2 to 10 HZ range. This implies that our calculations of ISRS are conservative. This is significant because other than a few tanks and yard equipment, safety-related systems and components are located at various elevations within structures and are designed and qualified to the calculated DBE and OBE ISRS.

The above assessment is based on the spectral peaks calculated from the recorded time-history data. The use of recorded data is conservative as determined in the NRC sponsored research documented in NUREG/CR-0098, prepared by N.M. Newmark and W.J. Hall. Section 3.1 of NUREG/CR-0098 states that, "Although peak values of ground motion may be assigned to the various magnitudes of earthquake, especially in the vicinity of the surface expression of a fault or at the epicenter, these motions are in general considerably greater than smaller motions which occur many more times in an earthquake. Design earthquake response spectra are based on "effective" values of the acceleration, velocity and displacement, which occur several times during the earthquake, rather than isolated peak values of instrumental reading. The effective earthquake hazards selected for determining design spectra may be as little as one-half the expected isolated peak instrument readings for near earthquakes, ranging up to the latter values for distant earthquakes."

When considering seismic capability of SSCs, equipment within containment, including the reactor core, internals, reactor coolant system piping, steam generators, and appurtenances, are designed to withstand the loads resulting from a combined design basis seismic and loss of coolant accident (LOCA) event. Because the LOCA, which was not experienced at North Anna, governs calculated loads for a combined seismic and LOCA event, there is additional assurance of the adequacy of design margins for the seismic-only loads experienced on August 23, 2011.

During the implementation of the Generic Letter (GL) 88-20, Supplement 4, *Individual Plant Examination of External Events* (IPEEE) and GL 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment In Operating Reactors (USI A-46),* programs, several modifications and improvements were made to enhance the plant's seismic safety. In the IPEEE effort, the plant was evaluated to a median-centered ground response spectrum shape anchored to 0.3g PGA. Calculations were performed to determine the high-confidence-of-low-probability-of-failure (HCLPF) capacities of equipment and structures, and only a small number of structures and components were found to have HCLPF capacities below 0.3g. The calculations for those components with HCLPF capacity below 0.3g are being reviewed for potential improvements to increase their seismic capacities; however, detailed inspections of these components by trained and experienced seismic review teams have shown no evidence of damage from the seismic event. Also, comprehensive inspections of plant SSCs were performed, and they likewise showed no significant physical or functional damage to safety-related plant SSCs and only limited damage to non-safety related, non-seismically designed SSCs. In addition, an inspection of plant areas and SSCs where earthquake induced damage would likely be most evident, including non-safety related structures and components such as large, unanchored water storage tanks, was performed by a seismic review team. The team, consisting of several Dominion engineers and industry experts, reached the same conclusion.

It should be noted that if the CAV value in one direction had been about 10% lower, it would be below the very conservative threshold of 0.16g-sec (the other two were already lower), the plant would not have had to shut-down because of this earthquake, provided Dominion had incorporated the RG 1.166 criteria for OBE exceedance into its licensing basis and seismic instrumentation design. Based on the non-damaging parameters (relatively small CAV and the limited effective strong motion duration) of the recent earthquake, detailed inspection results, surveillance and functional tests, improvements made during USI A-46 and IPEEE efforts, and inherent margins shown in the IPEEE program, it is concluded that North Anna's seismic design has adequate margins and the plant can continue to operate safely.

Regulatory Approach

Subject to the completion of the identified near-term actions, the regulatory evaluation and response demonstrates that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public, and provides the basis for restart of North Anna Units 1 and 2. Dominion's response framework was developed on the basis of the following guidance and related considerations: 10 CFR 100 Appendix A establishes a regulatory requirement following an event that exceeds the OBE earthquake. RG 1.167 provides NRC-endorsed guidance for licensee response to seismic events, and references the guidance of EPRI NP-6695 for required evaluations and inspections based on observed consequences of the seismic event. Inspections and surveillance and functional testing to confirm the functionality of plant SSCs are identified herein, and will be completed prior to restart. Dominion proposes the completion of these inspections and surveillance and functional tests as the necessary pre-conditions for North Anna restart, and requests NRC concurrence with this framework and proposed actions.

The August 23, 2011 event at North Anna is appropriately classified under the guidance provided in IAEA Safety Report Series No. 66, "Earthquake Preparedness and Response for Nuclear Power Plants," as Action Level 5, which requires initial focused inspections and testing prior to restart, and re-evaluation of the seismic hazard after restart, if deemed necessary. Guidance on earthquake response for this event classification requires no revision of the plant's seismic design prior to restart. Seismic design will continue to be evaluated in the context of response to NRC Generic Issue (GI) 199, *Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern US (CEUS) for Existing Plants*, and implementation of the NRC Near-term Task Force Recommendations identified in SECY-11-0124 dated September 9, 2011.

<u>Summary</u>

Dominion has concluded that the North Anna OBE and DBE spectral and peak ground accelerations were exceeded. However, previous evaluations for IPEEE and USI A-46 resolution, as well as the event specific, calculated values for CAV and effective strong motion duration, indicate that no significant damage should be found. The comprehensive walkdowns, inspections, evaluations and surveillances that have been completed confirm the expected lack of significant physical or functional damage to safety-related SSCs. In addition, the surveillance and functional tests and other identified items that will be completed prior to startup will confirm the ability of safety-related and plant support SSCs to perform their design basis functions. Finally, long-term actions have been identified to better inform and strengthen the capability of plant staff to promptly identify future earthquake intensity and to establish plans and methods for seismic evaluation of SSCs going forward pursuant to the resolution of NRC GI-199.

Therefore, it is Dominion's conclusion that, upon completion of the near-term activities discussed within this document, we will have demonstrated that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. NRC concurrence is hereby requested to restart North Anna Units 1 and 2 upon completion of the remaining near-term action items identified in Enclosure 8.

We also acknowledge receipt of the NRC request for additional information (RAI) dated September 14, 2011. Although much of the requested information is provided in this letter, we will provide a docketed response to the RAI in a timely manner.

If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

E. S. Grecheck
Vice President – Nuclear Development

Enclosures

Commitments made in this correspondence:

The commitments included in this correspondence are provided in Enclosures 8 and 9.

COMMONWEALTH OF VIRGINIA)

COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by E. S. Grecheck who is Vice President – Nuclear Development, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 17th day of September, 2011.

My Commission Expires: 4/30/2015.

Ginger Lynn Rutherford
NOTARY PUBLIC
Commonwealth of Virginia
Reg. # 310847
My Commission Expires 4/30/2015

Ginger L Rutherford
Notary Public

cc: U.S. Nuclear Regulatory Commission - Region II
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State Health Commissioner
Virginia Department of Health
James Madison Building - 7th floor
109 Governor Street, Suite 730
Richmond, Virginia 23219

Enclosure 1

Characterization of the North Anna Seismic Event of August 23, 2011

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station**

Characterization of the North Anna Seismic Event of August 23, 2011

The vibratory motions from the Magnitude 5.8 Mineral, Virginia earthquake of August 23, 2011 with its epicenter about 11 miles from the North Anna Power Station were recorded in three orientations (North-South [N-S], East-West [E-W], Vertical [V]) at several locations in the plant using two types of instruments: the Engdahl scratch plates that recorded 12 discrete spectral accelerations between 2 and 25.4 Hz, and the Kinemetrics analog recorders that recorded time-histories of the accelerations. The North Anna Updated Final Safety Analysis Report (UFSAR), Section 3.7.4.5, states, "Regulatory Guide 1.12 provides a general basis for selection of earthquake instrumentation for Seismic Class I structures and components where instruments are installed. The Guide specifies that the containment be instrumented with two triaxial acceleration sensors. Since the structure is founded on fresh rock, a separate "free field" triaxial acceleration sensor is not required." Consequently, since North Anna did not have free-field instruments, the most appropriate location to determine whether the plant Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE) were exceeded is at the top of the basemat of the containment structure. The containment was analyzed as a rock founded structure and the calculated spectra at the top of the basemat are close (but slightly conservative) to the free-field rock OBE and DBE spectra that are defined in the North Anna UFSAR. Therefore, this location was best suited for comparing the recorded motions to the North Anna design OBE and DBE spectra and to calculate the Cumulative Absolute Velocity (CAV) values, a damage parameter, from the recorded time-histories.

A comparison of the recordings from the Engdahl scratch plates at the containment basemat to OBE and DBE criteria (see Figures 2.1 and 2.2 in Attachment 2 to this Enclosure) showed that the plant OBE was exceeded. DBE criteria were slightly exceeded, but an analysis of data captured by the Kinemetrics recording was needed to make a final determination on whether DBE criteria had been exceeded. Dominion, together with several industry experts, considers the Kinemetrics analog recorder data to be more reliable. The Kinemetrics data shows that the OBE and DBE were exceeded in all three directions (see Figures 3.1 and 3.2 in Attachment 3); however, in the frequency range most damaging to equipment, 2 to 10 Hz, the DBE was not exceeded in the E-W direction, and the earthquake exceeded the DBE, on average, by about 12% in the N-S direction with the sharpest peak exceeding the corresponding DBE spectral ordinate by a factor of about 1.3, and about 21% in the vertical direction with the sharpest peak exceeding the DBE by a factor of about 1.6.

The CAV values, which are indicators of the earthquake's damage potential, were calculated from the recorded time-histories at the containment basemat. These values were below the very conservative threshold of 0.16g-sec defined in NRC Regulatory Guide (RG) 1.166 for OBE exceedance in the E-W and vertical directions, and the CAV limit was exceeded by about 10% in the N-S direction. Cumulative energy (Husid plots) calculations showed that the effective strong motion duration in the N-S direction was 1 second, in the vertical direction it was 1.5 seconds, and in the E-W direction it was 3.1 seconds. The CAV and the cumulative energy data are consistent with the findings from extensive plant inspections, including a specific plant inspection performed by a seismic review team consisting of Dominion engineers and several experts from the industry, which showed no significant physical or functional seismically induced damage to safety-related SSCs and only limited damage to non-safety related, non-seismically designed SSCs.

Based on the above, it is concluded that the August 23, 2011 Mineral, Virginia earthquake exceeded the spectral and peak ground accelerations for the OBE and DBE of the North Anna plant; however, CAV, which is an indicator of the earthquake damage potential, was marginally exceeded in only one direction and the effective strong motion duration of the earthquake was small.

North Anna Units 1 and 2 were reviewed in the 1990s under the Unresolved Safety Issue (USI) A-46 for design basis earthquake and under Generic Letter (GL) 88-20, Supplement 4, *Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities.* Numerous plant and procedural improvements and modifications were made during these efforts to improve the seismic safety of the plant. For the IPEEE, a seismic margin assessment was used with a review level earthquake of 0.3g peak ground acceleration (PGA) and a spectral shape given in NUREG/CR-0098. Only a small number of components (~50) were found to have a high-confidence-of-low-probability-of-failure (HCLPF) capacity less than 0.3g. These capacities are being reviewed for potential improvement. A thorough inspection of the small number of components with less than 0.3g HCLPF capacity (with the exception of one group where the relay functionality controlled the HCLPF capacity) is being performed by qualified and trained seismic review teams with no significant physical or functional damage detected to date. It is noted that the IPEEE spectra, based on NUREG/CR-0098 median-centered shape and anchored to 0.3g PGA, envelops the spectra at the containment basemat from the Kinemetrics recorders in all three directions.

With the exception of the IPEEE components with less than 0.3g HCLPF capacity that are currently being inspected, a detailed inspection of structures, systems and components (SSCs) was completed at North Anna and no significant physical or functional damage to safety-related plant SSCs has been identified and only limited damage to non-safety related, non-seismically designed SSCs has been found (See Enclosure 2). In accordance with the flow diagram in EPRI NP-6695 for EPRI Damage Intensity 1, post-shutdown inspections have been performed and surveillance and functional tests are being performed. Based on the inspection assessments, the EPRI plant damage intensity scale was characterized as "0" per EPRI NP-6695.

Several long term (post-startup) seismic evaluations and analyses will be performed for North Anna per EPRI NP-6695 and to address NRC Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern US (CEUS) for Existing Plants."

Earthquake Event Description

The US Geological Survey (USGS) reports that a Magnitude 5.8 earthquake occurred August 23, 2011 at 1351 hrs. The epicenter of this seismic event was reported to be at 37.936°N latitude, 77.933°W longitude, which places the event approximately 5 miles from Mineral, VA and 7 miles from Louisa, VA. Per reports, the epicenter was approximately 11 miles from the North Anna Power Station. The depth of the earthquake is reported to be 3.7 miles.

North Anna Power Station Response to the Earthquake

Both units at North Anna automatically tripped. The earthquake caused a series of reactor trip signals to both Unit 1 and Unit 2 reactors, as well as a total loss of offsite power to the station. Per the Event Review Report, the "First Out" reactor trip signals for both units were "High Flux Rate Reactor Trip". Other than the trip signals and subsequent loss of offsite power, which were either directly or indirectly caused by the seismic event, the plant responded as expected to the reactor trip. Separately, the 2H Emergency Diesel Generator developed a coolant leak and was subsequently manually secured.

North Anna Operating Basis and Design Basis Earthquake

North Anna UFSAR Section 3.7 provides the OBE and DBE ground response spectra and peak ground acceleration values. The DBE horizontal PGA for rock-founded structures is 0.12g and for soil founded structures it is 0.18g. Vertical direction PGAs are two-thirds of the horizontal accelerations and the OBE is half of DBE in the entire frequency range. The horizontal OBE and DBE curves for rock at 5% spectral damping are shown in Figure 1 and the horizontal OBE and DBE curves for soil at 5% spectral damping are shown in Figure 2.

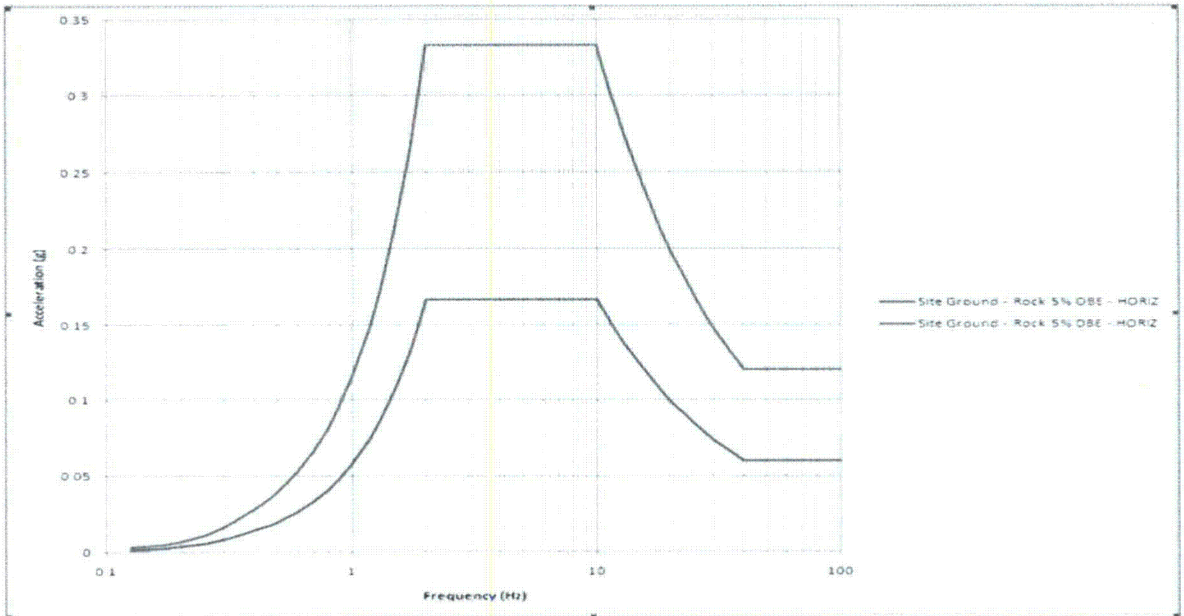


Figure 1 – Horizontal OBE and DBE Response Spectrum Curves for Rock at 5% Damping

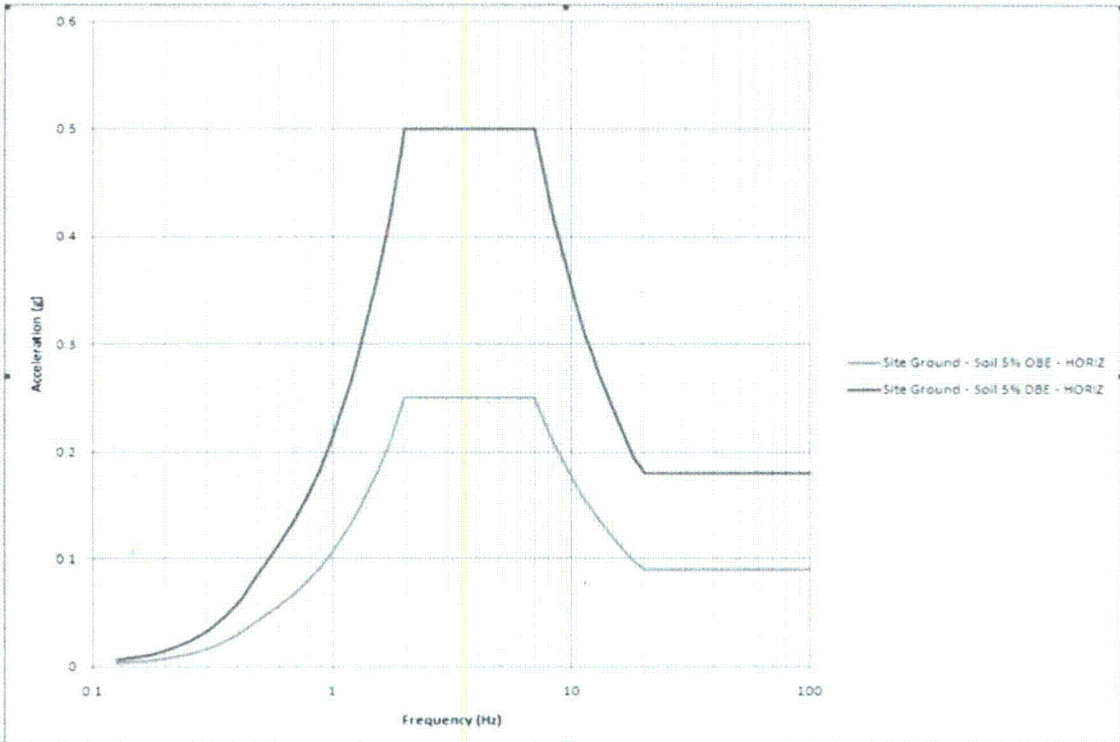


Figure 2 - Horizontal OBE and DBE Response Spectrum Curves for Soil at 5% Damping

NRC USI A-46 and IPEEE Programs and Plant Improvements

North Anna Units 1 and 2 were reviewed in the 1990s under USI A-46 for design basis earthquake and under the IPEEE program required by Generic Letter (GL) 88-20 Supplement 4 to determine vulnerabilities for a beyond design basis event. The USI A-46 program evaluated components in a safe shutdown equipment list (SSEL) that included 20 classes of equipment, tanks and heat exchangers, cable trays and conduits and relays. The IPEEE, which used a seismic margin assessment, consisted of an enhanced SSEL with two success paths to achieve safe shutdown. Several plant and procedural improvements and modifications were made during these efforts to improve the seismic safety of the plant. Some of the significant modifications included tying safety-related electrical cabinets together to prevent banging or relay chatter in the side to side direction for many rows of cabinets, modification of the anchorages of three tanks, improving anchorage of electrical cabinets and other components, improving the control room ceiling, improving seismic housekeeping and implementing a housekeeping procedure, reorienting valves to prevent higher stress along weak axis, and cable tray and conduit support improvements. For the IPEEE, a review level earthquake of 0.3g peak ground acceleration (PGA) was used. It is noted that in the IPEEE evaluations, only a small number of components were found to have a high-confidence-of-low-probability-of-failure (HCLPF) capacity less than 0.3g. These components are listed in Table 1 below. Further, a few masonry block walls were also reported during IPEEE with a capacity less than 0.3g. These block walls are listed in Table 2. A thorough inspection of these components is being performed by qualified and trained seismic review teams and no significant physical or functional damage has been reported to date. The capacities of these components will be reviewed for potential improvements.

Based on the improvements made to the plant during the USI A-46 and the IPEEE programs, the plant has substantial seismic margin over its initial design. In addition, after USI A-46 was completed, North Anna's procedure for seismic qualification of equipment was updated to use the later versions of IEEE standard 344 (1975 or 1987 version) for seismic qualification of equipment, which also provides improved safety margins.

The rock and soil free-field median-centered response spectra, based on NUREG/CR-0098 and anchored to a 0.3g PGA, were used as the Review Level Earthquake for IPEEE. These spectra are shown in Figure 3.

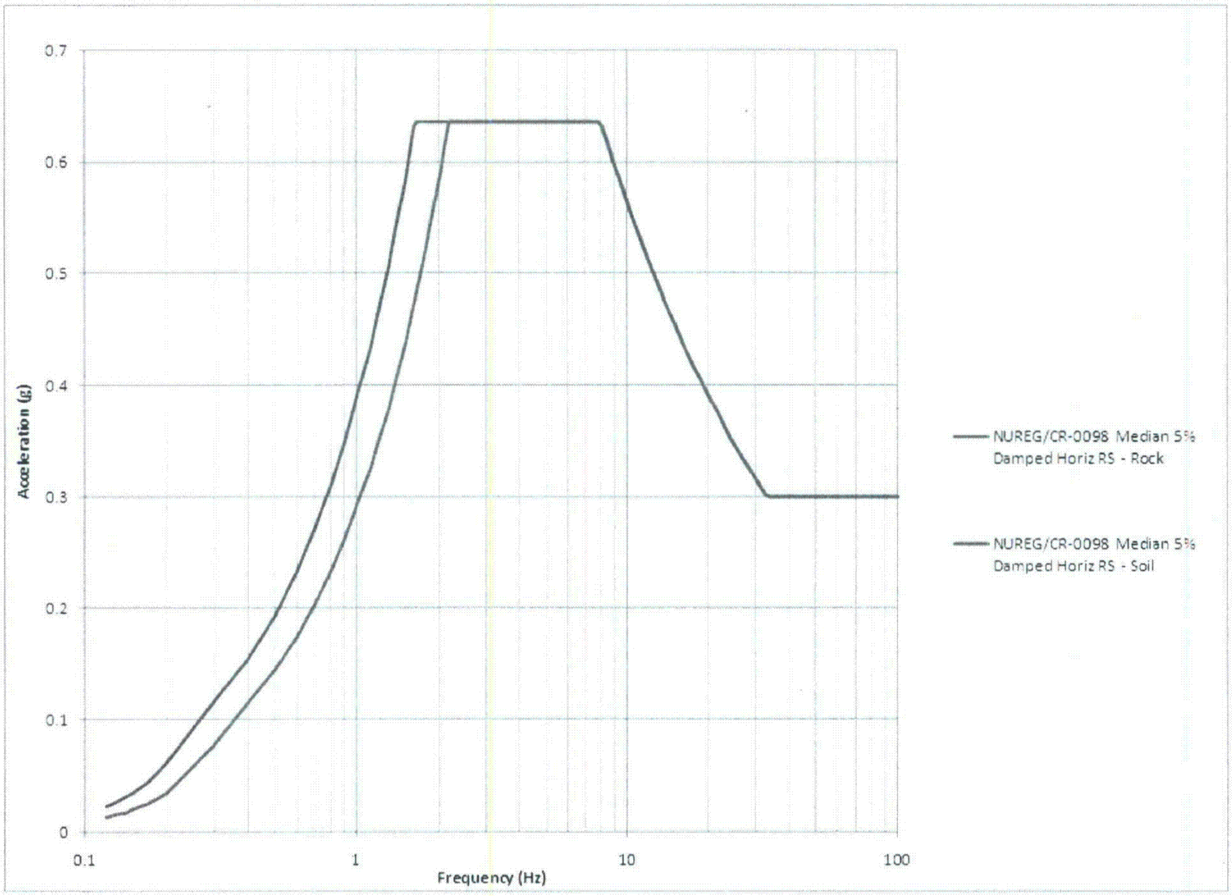


Figure 3 - Review Level Earthquake used in IPEEE – Rock & Soil Founded Structures – Horizontal Direction, 5% Spectral Damping

Table 1 - Summary of HCLPF Capacities Less than 0.3g (IPEEE)

Equipment Mark Number	Equipment Description	HCLPF Capacity	Mode of Failure
1(2)-CN-TK-1	*Emergency Condensate Storage Tanks*	0.16 g	*Overturning moment capacity – see note below*
1(2)-QS-TK-1	Refueling Water Storage Tanks	0.18 g	Overturning moment capacity
1-EP-CB-04A, B, C, D 2-EP-CB-04A, B, C, D	120 V Vital AC Bus	0.19 g	Anchorage
1-QS-TK-2	Refueling Water Chemical Addition Tank – Unit 1	0.19 g	Foundation Overturning
1-CH-TK-1A, B, C	Boric Acid Tanks	0.21 g	Anchorage
1-HV-AC-1, 2 2-HV-AC-8, 9	Control Room Air Conditioners	0.21 g	Anchorage
1(2)-EI-CB-21	Sequence of Events Recorders	0.22 g	Anchorage
1-EE-SW-1H, 1J 2-EE-SW-2H, 2J	4160 V Emergency Bus	0.23 g	Relay Capacity
2-QS-TK-2	Refueling Water Chemical Addition Tank – Unit 2	0.24 g	Foundation Overturning
2-EE-BKR-RTA, RTB, BYA, BYB (Cabinets 2-EI-CB-46A, B)	Reactor Trip Breakers –(Unit 2 only)	0.24 g	Anchorage
1(2)-HV-E-4A, B, C	Heating and Ventilation Chiller Units	0.27 g	Anchorage
1-BD-TV-100A, 100C, 100E 2-BD-TV-100A, 100C, 100F	SG Blowdown Containment Isolation Valves	0.28 g	Cast Iron Yokes
1(2)-CC-P-1A, 1B	Component Cooling Water Pumps	0.29 g	Anchorage

Note: The tank is unanchored; however, the tank is enclosed in a concrete missile shield which is approximately 2" away from the tank. This 2" space is filled with Rotofoam. The Rotofoam and the concrete missile shield were not considered when calculating the HCLPF.

Table 2 - Block Walls with HCLPF Capacity Less than 0.3g (IPEEE)

Group	Walls	Construction	Class	Bounding Wall	SMA HCLPF (g)
5	SB-271-17 SB-271-85 SB-294-3 Turbine Building Walls	8" and 12" Thickness	III	SB-271-17	0.21

North Anna Seismic Instrumentation

Refer to Attachment 1.

Engdahl Scratch Plate Recorder Data and Comparisons to Design Basis

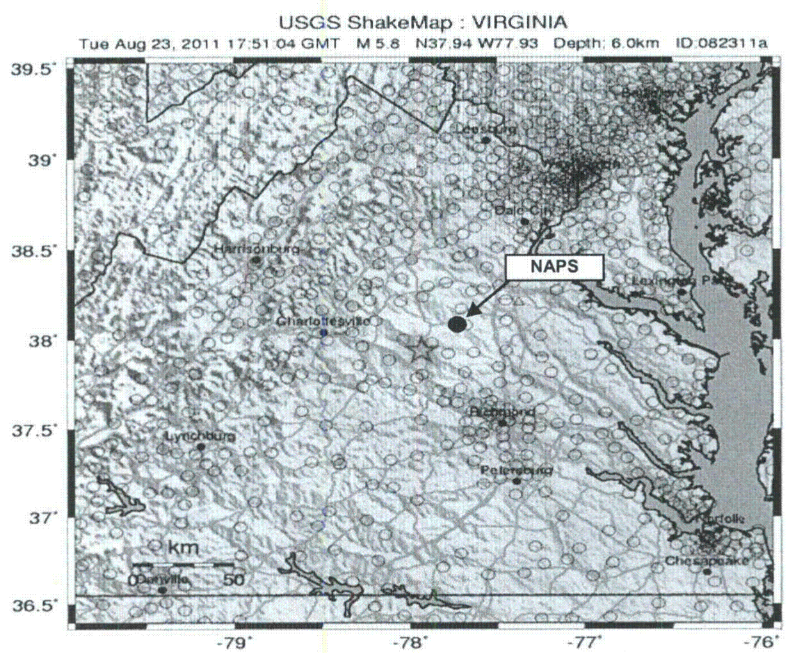
Refer to Attachment 2.

Kinemetrics Triaxial Recorder Data and Comparisons to Design Basis and IPEEE

Refer to Attachment 3.

Earthquake Characterization Based on Data from External Sources

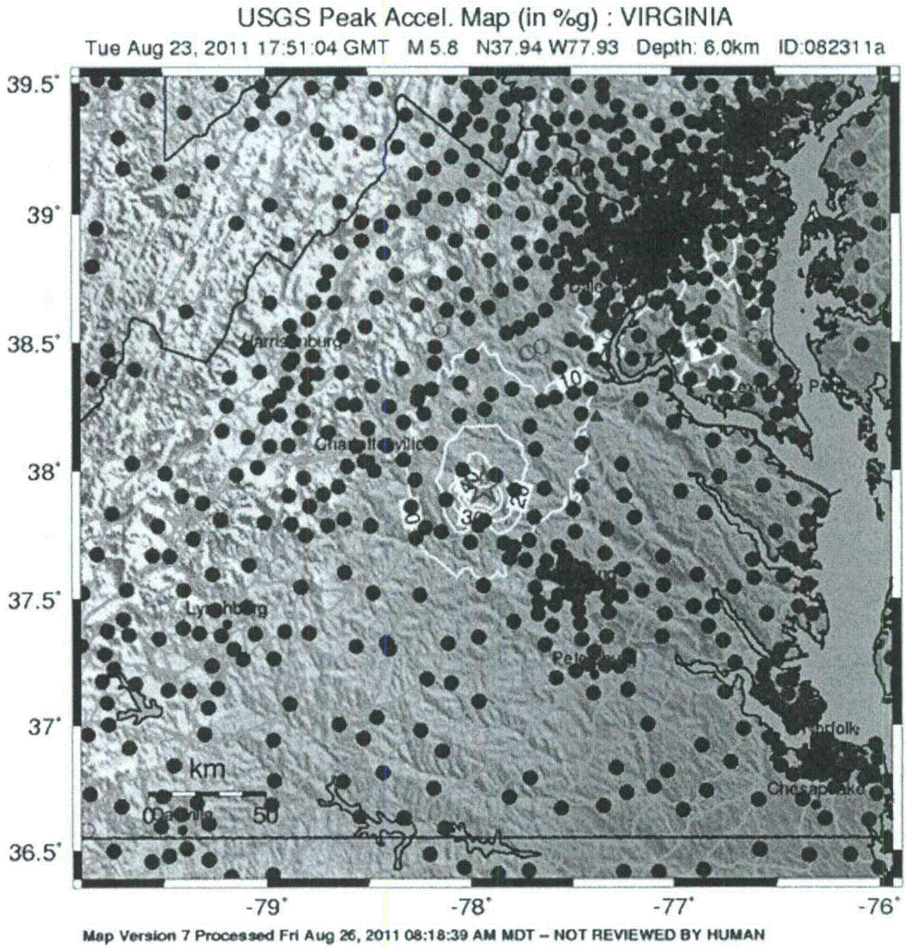
The following data about the August 23, 2011 Mineral, Virginia earthquake was obtained from publicly available information from the website of the United States Geological Survey (USGS). North Anna Power Station is approximately 11 miles to the northeast of the epicenter (shown approximately on the map below).



Map Version 7 Processed Fri Aug 26, 2011 08:18:39 AM MDT -- NOT REVIEWED BY HUMAN

PERCEIVED SHAKING	Not felt	Weak	Light	Moderate	Strong	Very strong	Severe	Violent	Extreme
POTENTIAL DAMAGE	none	none	none	Very light	Light	Moderate	Moderate/Heavy	Heavy	Very Heavy
PEAK ACC.(%g)	<.17	.17-1.4	1.4-3.9	3.9-9.2	9.2-18	18-34	34-65	65-124	>124
PEAK VEL.(cm/s)	<0.1	0.1-1.1	1.1-3.4	3.4-8.1	8.1-16	16-31	31-60	60-116	>116
INSTRUMENTAL INTENSITY	I	II-III	IV	V	VI	VII	VIII	IX	X+

Source: http://earthquake.usgs.gov/earthquakes/shakemap/global/shake/082311a/



Source: http://earthquake.usgs.gov/earthquakes/shakemap/global/shake/082311a/download/pga.jpg

Based on the above PGA shake map, the location of the plant is just outside of the 0.3g contour plotted above. This appears to be consistent with the peak ground acceleration measured by the Kinemetrics recorder at the Containment basemat (see Attachment 3).

Interpretations of the Recorded Data Compared to North Anna Plant Design Basis

An assessment of the Engdahl scratch plate recorded data is shown in Attachment 2, and similarly, the assessment of the Kinemetrics data is shown in Attachment 3. These attachments include the comparison plots at various locations between the recorded data and the North Anna OBE and DBE spectra calculated from time-history dynamic analyses of the two structures in which these recorders are located. The most relevant recorded data is from the Kinemetrics recorders at the basemat of the containment structure. Based on our discussions with Engdahl and with several industry experts, we believe that the Kinemetrics data is more reliable than the Engdahl data at the same location. As discussed in Attachment 3, the Kinemetrics recorders show that the North Anna OBE and DBE spectral and peak ground accelerations were exceeded.

Figure 3.1 in Attachment 3 also compares the response spectra created from the Kinematrics data in the horizontal directions to the Containment basemat spectra used for IPEEE, which is considered close to the free-field rock. The IPEEE curves envelop the spectra from the recorded data for the Containment basemat elevation. Similarly, Figure 3.2 in Attachment 3 shows that for the vertical direction the IPEEE also envelops the recorded data.

Criteria from EPRI NP-6695 and RG 1.166, and North Anna Data from the August 23, 2011 Event

EPRI NP-6695 provides basically two criteria to determine OBE exceedance, either one of which could be used. One of these two criteria is that the Cumulative Absolute Velocity (CAV) should be less than 0.3g-sec. Subsequent to the initial issue of NP-6695, the methods and the criterion to calculate the CAV were updated (EPRI Report TR-100082) to remove the effects on the CAV of very small cycles of motion. The updated limit was conservatively set at 0.16g-sec. RG 1.166 uses this new CAV criterion of 0.16 g-sec. The CAV values from the time-histories of the August 23, 2011 event at the containment basemat were calculated by three independent consultants in each of the three directions. All three consultants calculated approximately the same values, as shown in Table 3 below. It is noted that per EPRI NP-6695 and RG 1.166, although CAV values should be calculated from the free-field recorded data, it was judged that the top of the containment mat would give a reasonable approximation of the free-field, because the containment is rock-founded and the calculated spectra at the top of the containment mat is fairly close to the corresponding OBE or DBE free-field spectra. The analysis of the recorded data shows that the CAV values were below the very conservative threshold of 0.16g-sec defined in NRC RG 1.166 for OBE exceedance in the E-W and vertical directions, and the CAV exceeded the limit by about 10% in the N-S direction. The cumulative energy (Husid) plots were developed by Simpson, Gumpertz & Heger (SGH) from the containment mat recorded time-histories. These plots show that the effective strong motion duration in the N-S direction was 1.0 second, in the vertical direction it was 1.5 seconds, and in the E-W direction it was 3.1 seconds. The CAV and cumulative energy data are consistent with the inspections, including a specific plant inspection performed by a seismic review team consisting of Dominion engineers and several experts from the industry, which showed no significant physical or functional seismically induced damage to non-safety or safety-related structures and components.

Table 3 - Cumulative Absolute Velocity Results

	East-West (g-sec)	North-South (g-sec)	Vertical (g-sec)
Kinemetrics	0.137	0.175	0.118
SGH	0.118	0.169	0.105
Bechtel (preliminary)	0.134	0.181	0.113
Average	0.130	0.175	0.112

Conclusions on Characterization of the North Anna Earthquake Based on the Review of Data

Based on the recorded plant Kinematrics data, as documented in Attachment 3, it is concluded that the August 23, 2011 earthquake exceeded the spectral and peak ground accelerations for the OBE and DBE of North Anna plant; however, CAV, which is an indicator of the earthquake damage potential was only marginally exceeded in only one direction and the strong motion duration of the earthquake was small based on the recorded and calculated parameters.

Conclusions Based on Comprehensive Plant Inspections

Comprehensive inspections of non-safety and safety related SSCs were performed. These inspections are discussed in Enclosure 2. Extensive testing of safety related systems and components is ongoing and will be completed prior to each respective unit's startup. The inspections did not reveal any significant physical or functional damage or deformation of safety related SSCs. Reported damage and observations from the earthquake include spurious actuation of the sudden pressure relays on the Reserve Station Service Transformers, limited cracking of ceramic/porcelain components on switchyard equipment and limited cracking of non-safety walls, and movement of the Independent Spent Fuel Storage Installation (ISFSI) casks. The most significant visual damage of a non-safety related SSC was spalled concrete on a condensate polishing tank support pedestal that did not affect function. The most significant damage that required repair on non-safety related equipment was the Generator Step Up (GSU) transformer bushing leakage. This is a heavy bushing which projects out from the transformer and is primarily supported on one end; thus, this damage is not surprising. Other than the above, no significant physical or functional seismically induced damage to non-safety related structures or components has been identified.

Attachment 4 and Enclosure 2 discuss the in-process detailed inspections of the low capacity components identified in IPEEE by trained and experienced seismic review teams that, to date, have not identified any significant physical or functional earthquake based damage or anomaly for these components. Based on the results of the inspections conducted, the earthquake damage to North Anna Power Station is characterized as EPRI Damage Scale Intensity "0" per NP-6695. Based on these observations and the fact that the earthquake maximum CAV value is 0.175 (based on an average of the three values provided in Table 3), no significant physical or functional damage of safety related components would be expected.

Separately, a limited scope plant inspection was conducted by a Seismic Review Team that included Dominion and industry seismic experts. No significant physical or functional earthquake-induced damage was observed for the areas and SSCs inspected. The results of this plant inspection are summarized in Attachment 4 to this Enclosure.

Subsequent to the August 23, 2011 earthquake, USGS has recorded twenty-four aftershocks from August 23 to September 1, 2011 ranging from Magnitude 1.8 to 4.5.

Conclusions

Based on the recorded plant data, it is concluded that the Mineral, Virginia earthquake of August 23, 2011 exceeded the spectral and peak ground accelerations for the OBE and DBE of the North Anna plant; however, CAV, which is an indicator of the earthquake damage potential marginally exceeded the criterion in RG 1.166 in one direction only and the strong motion portion of the earthquake was of relatively small duration based on the recorded and calculated parameters.

Further, comprehensive plant inspections have concluded that the damage to the plant was minimal. Based on the site inspections, documented in Enclosure 2 that were conducted in accordance with EPRI NP-6695, the seismic damage is classified as Damage Intensity 0.

It is concluded that while the recorded data indicating exceedance of the OBE and DBE acceleration spectra, analysis of the time history of the spectra indicates that this earthquake should not have caused significant physical or functional damage to North Anna. The findings of no significant physical or functional damage to safety related SSCs and only limited damage to non-safety related SSCs are supported by the physical inspections and data reflecting the very low damage potential of the earthquake at the site (CAV and Husid plots).

North Anna Seismic Instrumentation

Overview of North Anna Seismic Instrumentation

A brief overview of the North Anna Power Station seismic event recording instrumentation is provided. The power station has two separate recording systems, one provided by Kinemetrics Inc. and the other provided by Engdahl. Both systems provide input to the Main Control Room via a common instrumentation panel located on the Backboards of the Unit 2 side. The panel mark number is 1-EI-CB-151. The Kinemetrics system will be discussed first followed by Engdahl.

[1] Kinemetrics

The sensors for this system are located inside Unit 1 Containment (CTMT). The sensor locations and types are as follows:

Sensor Mark #	Sensor Type	Location/Elevation	Equipment Mounting
1-ER-YE-01	Triaxial Time History Accelerograph, FBA-3	Unit 1 Containment 216'	U1 CTMT Mat
1-ER-VBS-101	Triaxial Seismic Trigger, TS-3	Unit 1 Containment 216'	U1 CTMT Mat
1-ER-VBS-102	Triaxial Seismic Switch, SP-1/TS-3	Unit 1 Containment 216'	U1 CTMT Mat
1-ER-YE-02	Triaxial Time History Accelerograph, FBA-3	Unit 1 Containment 291'	U1 CTMT Operating Deck

Seismic Trigger (TS-3), 1-ER-VB-101, activates at a sensed acceleration of 0.01g in any direction. The trigger starts the tape recorders for the CTMT Mat and Operating deck to record a time history of the event. It also initiates the event indicator (turns from black to white), local event alarm (yellow light) on SMA-3 control panel, and Earthquake Trouble Annunciator, window 1A-B6, on the Main Control Board. The trigger does not lock in and will reset on its own once the seismic event ends. A time delay ensures that the tape recorders continue recording for 10 seconds after the trigger has returned to normal. Once the trigger resets, the Main Control Board annunciator and local event alarm will clear. The event indicator requires a manual reset.

Seismic Switch (SP-1/TS-3), 1-ER-VBS-102, activates at a sensed acceleration of 0.04g vertical and 0.06g horizontal. Switch activation initiates the Earthquake Trouble annunciator, 1A-B6, and a yellow event alarm on the seismic switch power supply drawer. Both auto reset after the switch resets.

Kinemetrics Specifications:

Tape recorders (SMA-3 and FBA-3)	Scaled for ± 1g	Response 0 to 50Hz
Trigger (TS-3)	Adjustable from 0.005 to 0.05g	Flat response 1 to 10Hz
Switch (SP-1/TS-3)	Adjustable from 0.025 to 0.25g	Flat response 0.5 to 15Hz

North Anna Seismic Instrumentation

Kinemetrics is an active system and requires power. The Main Control Room panel receives 120VAC power from Semi-Vital Bus "1A." This panel loses power during a loss of offsite power (LOOP) until the 1H Emergency Diesel Generator (EDG) picks up the load on the bus. This panel will receive power as soon as the EDG output breaker closes as this load rides the bus. Kinemetrics instrumentation is provided with a battery back-up in control room panel 1-EI-CB-151 that is sized to provide one hour of back-up power. This ensures that the local alarms activate and that the tape recorders record the seismic event even with a LOOP. The input relays to the Main Control Board Earthquake Trouble Annunciator are Westinghouse ARD control relays that do not have battery back-up. With a LOOP, the Earthquake Trouble Annunciator will not function.

[2] Engdahl

Sensors for this system are located in the Unit 1 Containment and in the Auxiliary Building. The sensor locations and types are as follows:

Sensor Mark #	Sensor Type	Location/Elevation	Equipment Mounting
1-ER-RCDR-216A/B/C	Triaxial Response Spectrum Recorder PSR-1200	Unit 1 Containment 216'	Unit 1 CTMT Mat
1-ER-RCDR-231A/B/C	Triaxial Response Spectrum Recorder PSR-1200	Unit 1 Containment 231'	Unit 1 Residual Heat Removal (RHR) pump and heat exchanger area
1-ER-RCDR-244A/B/C	Triaxial Response Spectrum Recorder PSR-1200	Aux Building 244'	In between Unit 1 and 2 Component Cooling (CC) pumps
1-ER-RCDR-274A/B/C	Triaxial Response Spectrum Recorder PSR-1200	Aux Building 274'	Near Unit 1 "A" CC heat exchanger, 1-CC-E-1A
1-ER-RCDR-03	Triaxial Peak Accelerograph PAR-400	Unit 1 CTMT 218'	On pipe 12"-SI-125-1502, Unit 1 "C" Safety Injection Accumulator discharge piping
1-ER-RCDR-02	Triaxial Peak Accelerograph PAR-400	Unit 1 CTMT 241'	On Unit 1 "B" RHR heat exchanger, 1-RH-E-1B
1-ER-RCDR-04	Triaxial Peak Accelerograph PAR-400	Aux Building 279'	On Unit 1 "A" CC heat exchanger, 1-CC-E-1A

These devices are passive and have no immediate earthquake output except for 1-ER-RCDR-216A/B/C, which provides input to the Main Control Room Peak Shock Annunciator panel located in 1-EI-CB-151.

North Anna Seismic Instrumentation

Engdahl Specifications:

PAR-400 scribe plates	0 to 5g	0 to 51Hz, tri-axis, one plate in each axis
PSR-1200	0 to 2g	2 to 25.4Hz in 1/3 octave increments, tri-axis, 12 plates in each axis

Engdahl is a passive system and requires no power except for the Peak Shock Annunciator (PSA) in the Main Control Room. The PSA panel is located in 1-EI-CB-151 and is powered from Semi-Vital bus same as Kinemetrics. The PSA panel; however, does not have back-up power. During a LOOP, this panel does not function. If a seismic alarm came in during the period of time between the loss of power and cleared before restoration of power; the alarm circuits will not indicate this alarm on restoration of power.

Kinemetrics Measuring Equipment Discrepancies

It was identified that the Kinemetrics Seismic Instrumentation, FBA-3 Time History Recorders, were incorrectly installed. Specifically, devices 1-ER-YE-01 and 1-ER-YE-02 were 90 degrees off. The CTMT operating level (291') accelerometers cable was pointing called West and the CTMT basement was pointing called East. Per the vendor and the vendor tech manual, the cable coming out should be pointing towards called North. Therefore, the results obtained for the Transverse direction, which are supposed to correlate to East-West, apply to the North-South direction and the results obtained for the longitudinal direction, which are supposed to correlate to North-South, apply to the East-West direction.

Transverse = North-South
Longitudinal = East-West

This error in installation was relayed to Kinemetrics via Dominion prior to processing of the data; therefore, the Kinemetrics results properly account for the discrepancy.

Note that a subsequent concern was raised regarding the possibility that the East-West and Vertical Direction were errantly switched as well, based on the comparisons of the recorded spectra at the basemat and elevation 291' of the containment. An investigation was undertaken to determine whether there were any additional wiring/configuration errors. No additional anomalies were identified.

Engdahl Scratch Plate Data

Review and Comparison of the Data

The scratch plates retrieved from the Engdahl PSR-1200 scratch recorder units were sent to the vendor for reading. The Engdahl data consists of measurements at 2% damping for 12 discrete frequencies ranging from approximately 2 to 25 Hz.

Results are plotted below for the 4 locations where these recorders are installed:

1. RC Containment base mat elevation – Figures 2.1 and 2.2 in this Attachment
2. RC Containment RHR Flats elevation (231') – Figures 2.3 and 2.4 in this Attachment
3. Auxiliary Building base mat elevation (244') – Figures 2.5, 2.6, and 2.7 in this Attachment
4. Auxiliary Building elevation 273' near the CCHXs – Figures 2.8, 2.9, and 2.10 in this Attachment

The above plots of the Engdahl recorded accelerations include a comparison of the data to design OBE and DBE spectra for the corresponding Auxiliary and Containment elevations at 2% spectral damping.

As previously documented in this enclosure, it is the consensus of Dominion engineering and Dominion's consultants that the recorded data from the Kinemetrics instruments (refer to Attachment 3), located on the basemat of the Unit 1 Containment Building at North Anna plant reflects the best source of earthquake time-histories and response spectra to determine whether the OBE and DBE at North Anna plant were exceeded. The Engdahl data, by comparison, is considered to be less reliable and concerns were raised over the adequacy of the recorded data based on differences between Engdahl and Kinemetrics results and also the existence of several "zero" readings obtained at certain frequencies.

Containment Basemat Elevation

As demonstrated in Figure 2.1 for horizontal directions and Figure 2.2 for vertical directions, the Engdahl data for the Containment basemat elevation shows that OBE levels were exceeded. For DBE, however, there are only slight exceedances at a few frequencies.

Engdahl Scratch Plate Data

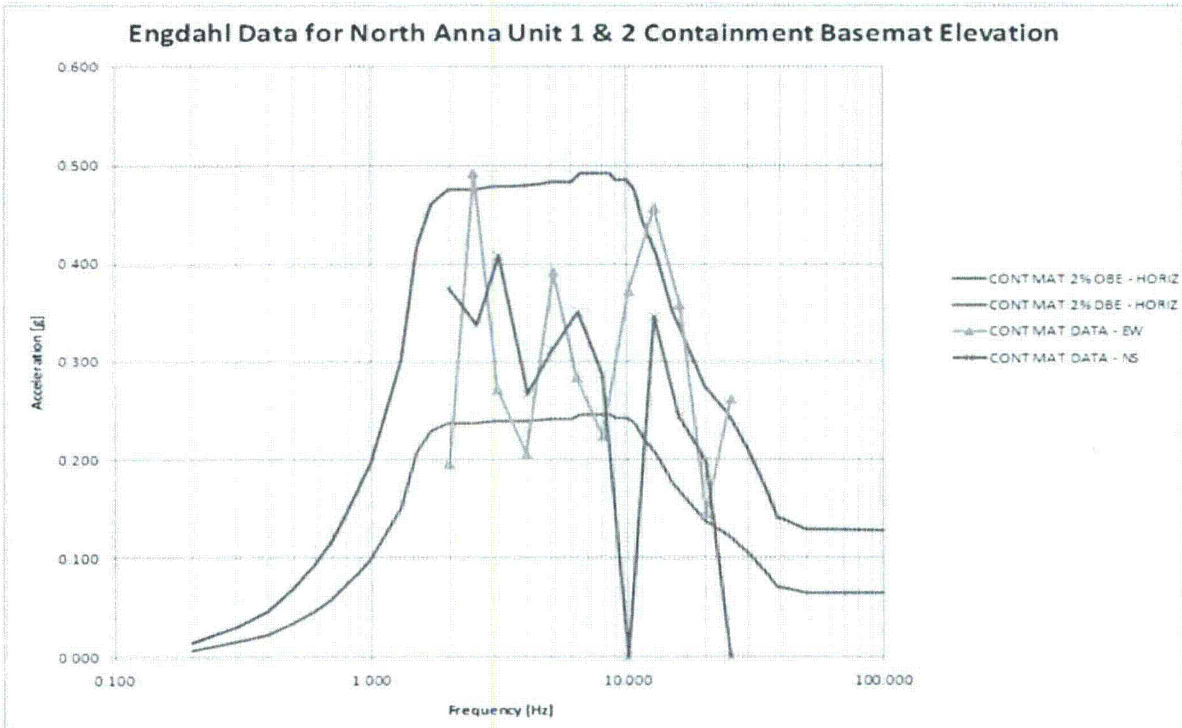


Figure 2.1 – Engdahl Data from Containment Basemat Recorder – Horizontal Directions

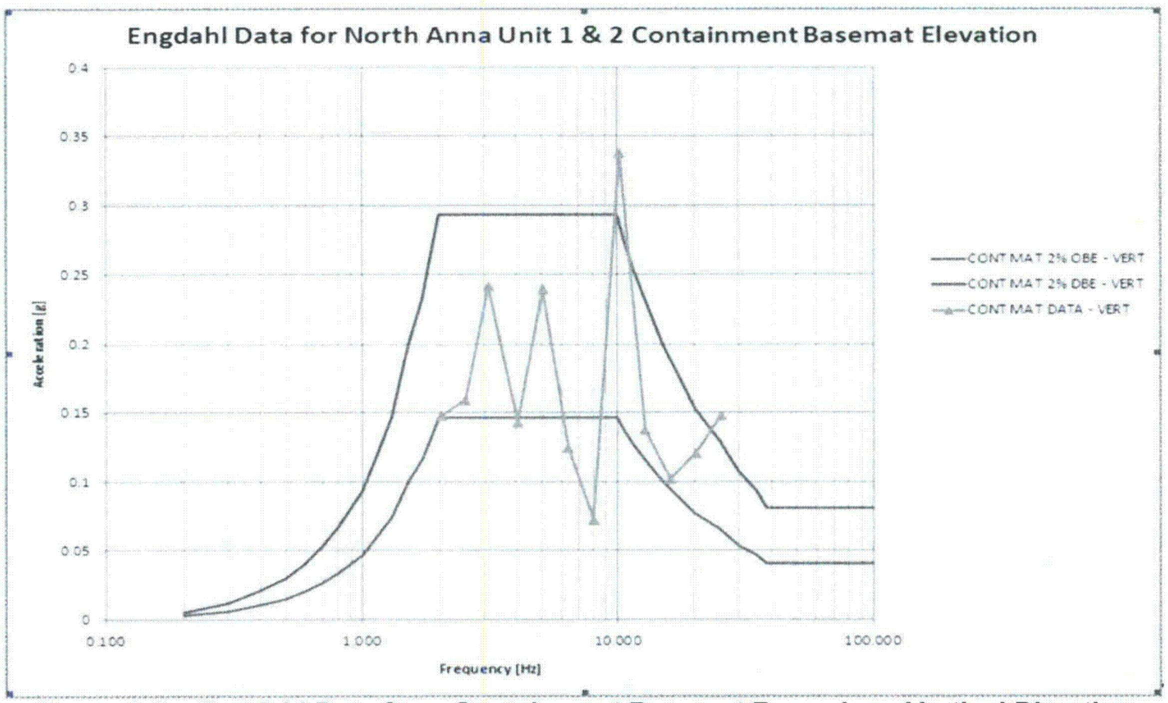


Figure 2.2 – Engdahl Data from Containment Basemat Recorder – Vertical Direction

Engdahl Scratch Plate Data

Containment RHR Flat Elevation (231')

As demonstrated in Figure 2.3, Engdahl recorded data in East-West and North-South directions at the RHR Flat elevation of Containment are less than OBE and DBE for frequency ranges up to approximately 16 Hz. Exceedances over DBE in the East-West direction occur around 25 Hz, in the high frequency range.

As demonstrated in Figure 2.4, Engdahl recorded data in vertical direction at the RHR Flat elevation of Containment exceeds OBE in several frequencies, but remains less than DBE.

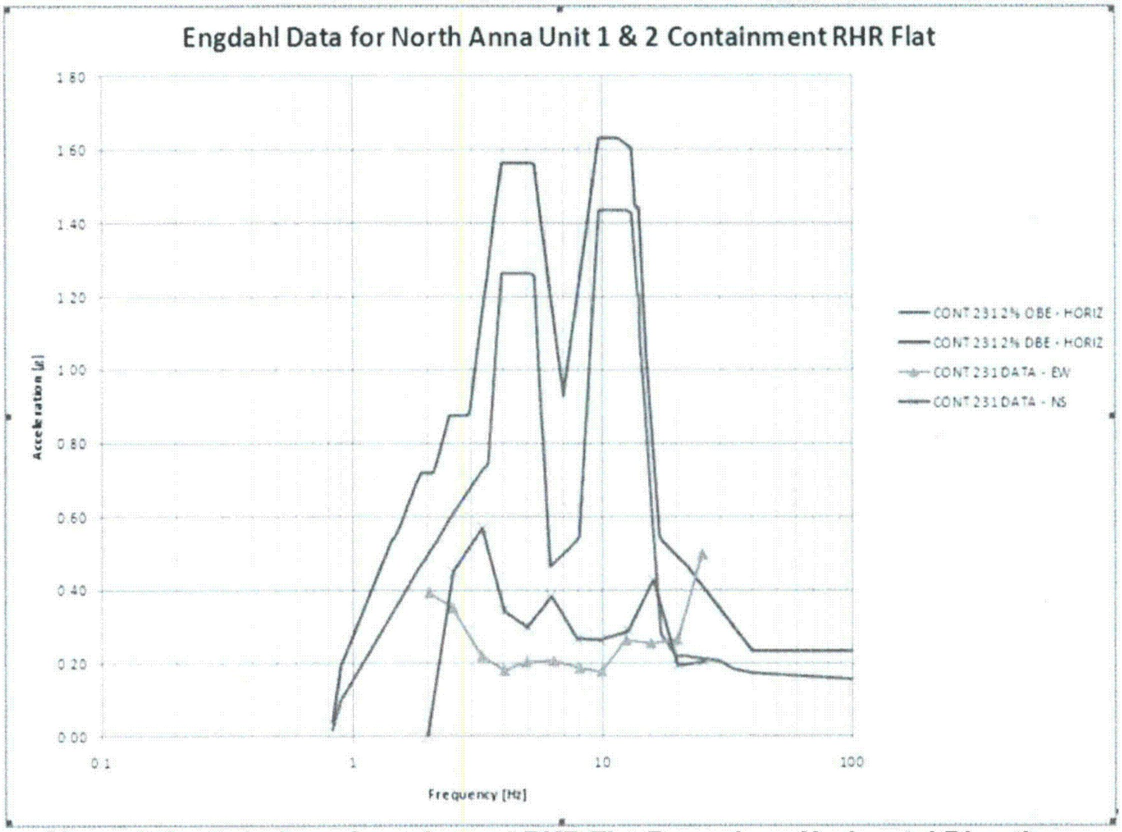


Figure 2.3 – Data from Containment RHR Flat Recorder – Horizontal Directions

Engdahl Scratch Plate Data

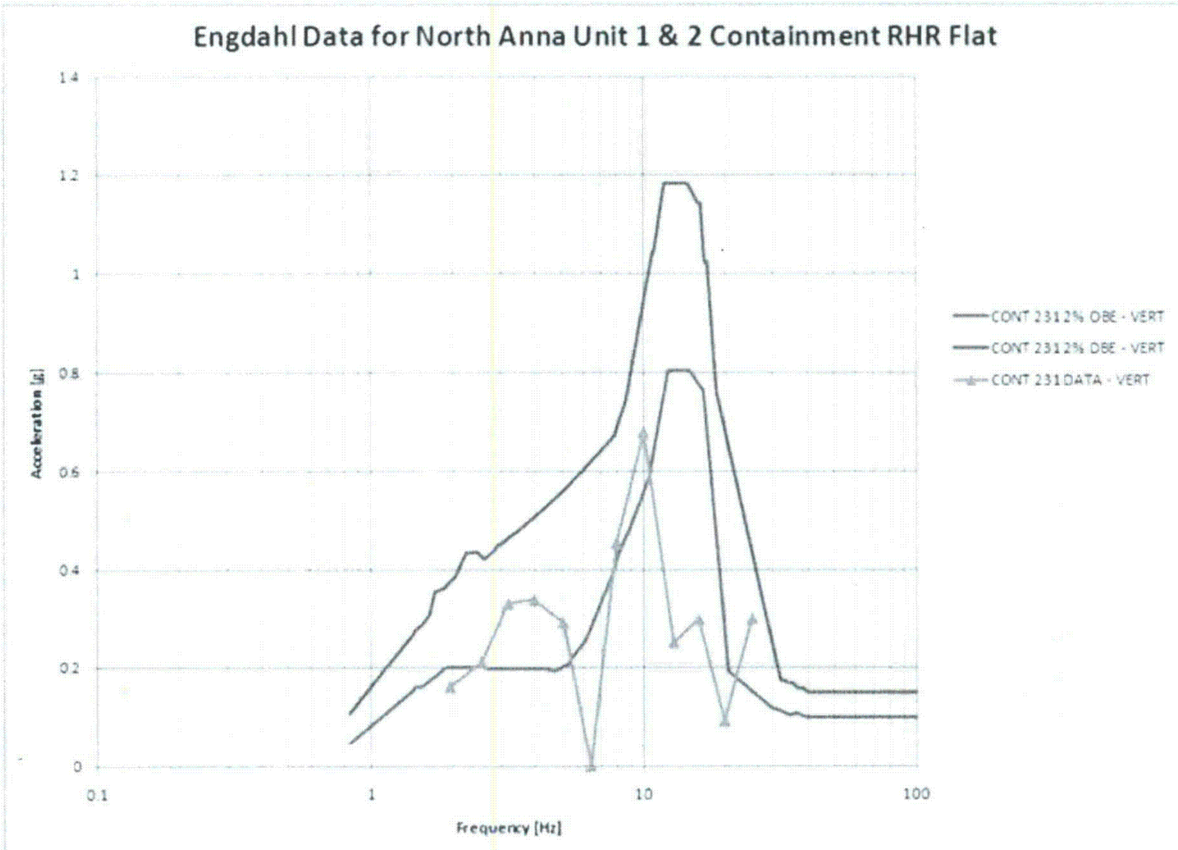


Figure 2.4 – Engdahl Data from Containment RHR Flat Recorder – Vertical Direction

Engdahl Scratch Plate Data

Auxiliary Building 244' and 273' Elevations

Figure 2.5 – Engdahl Data from Aux Building 244' Recorder – E-W Direction

Engdahl Scratch Plate Data

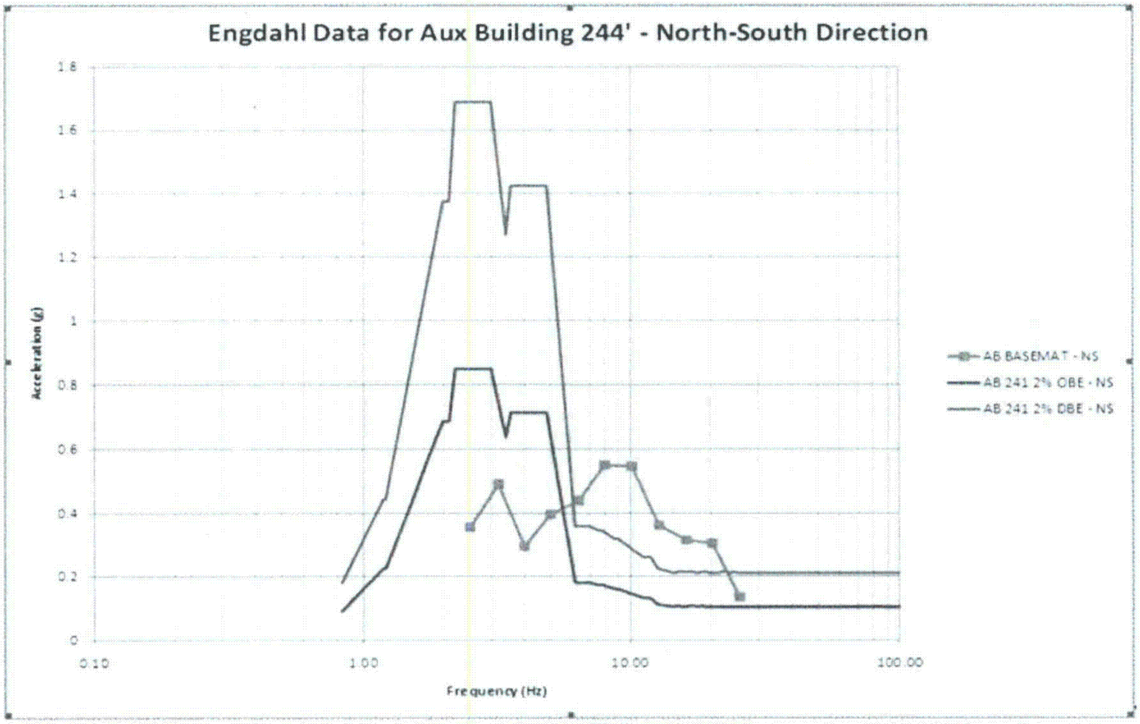


Figure 2.6 – Engdahl Data from Aux Building 244' Recorder – N-S Direction

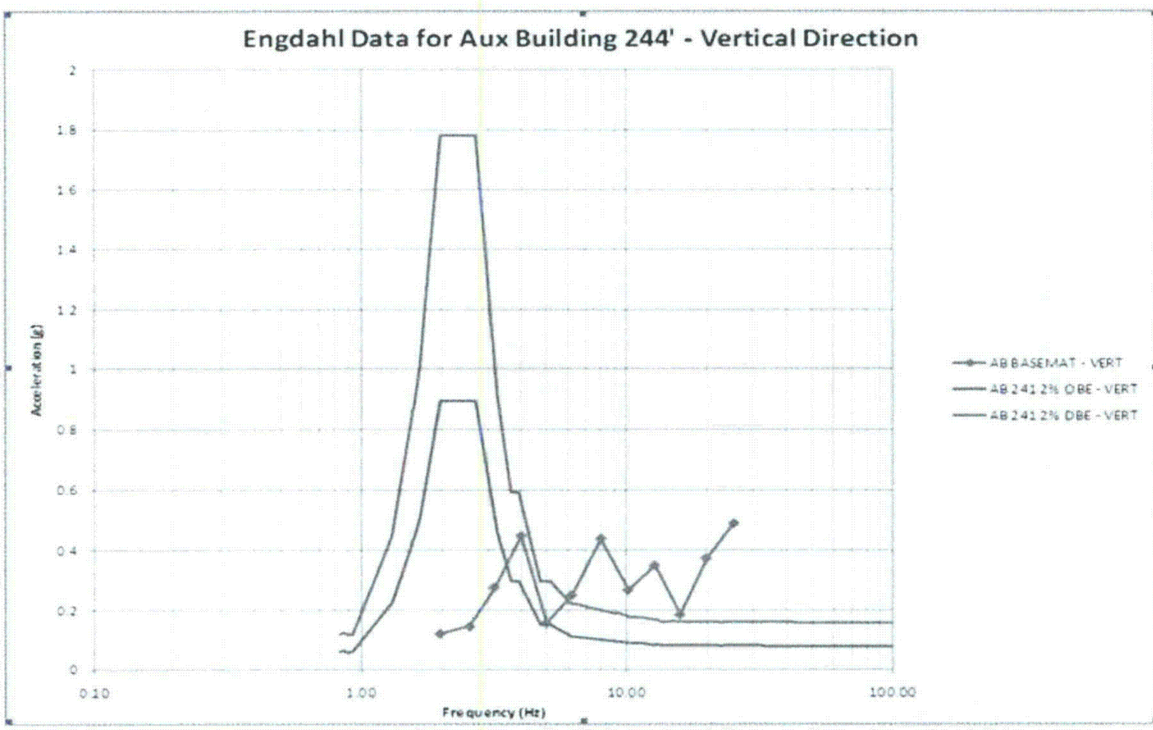


Figure 2.7 – Engdahl Data from Aux Building 244' Recorder – Vertical Direction

Engdahl Scratch Plate Data

Auxiliary Building 273' Elevation

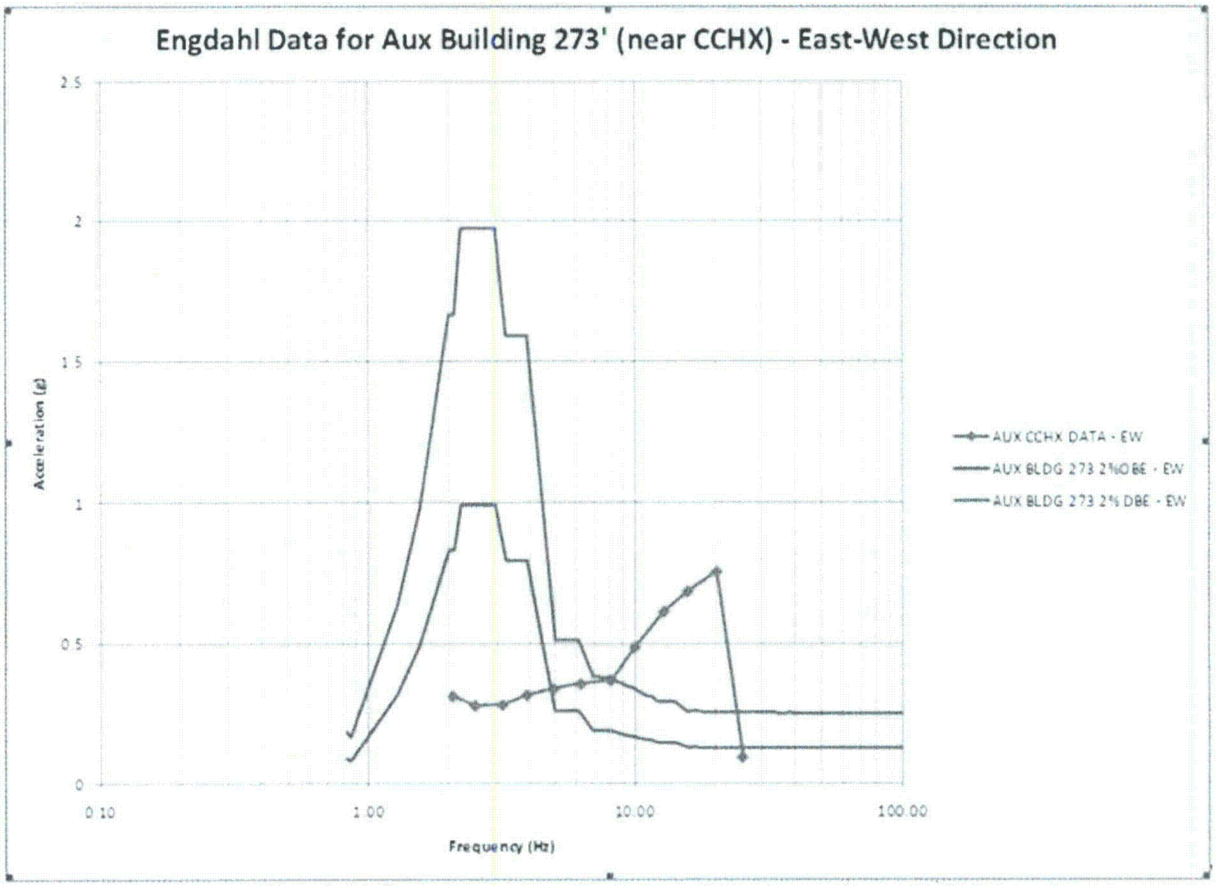


Figure 2.8 – Engdahl Data from Aux Building 273' Recorder – E-W Direction

Engdahl Scratch Plate Data

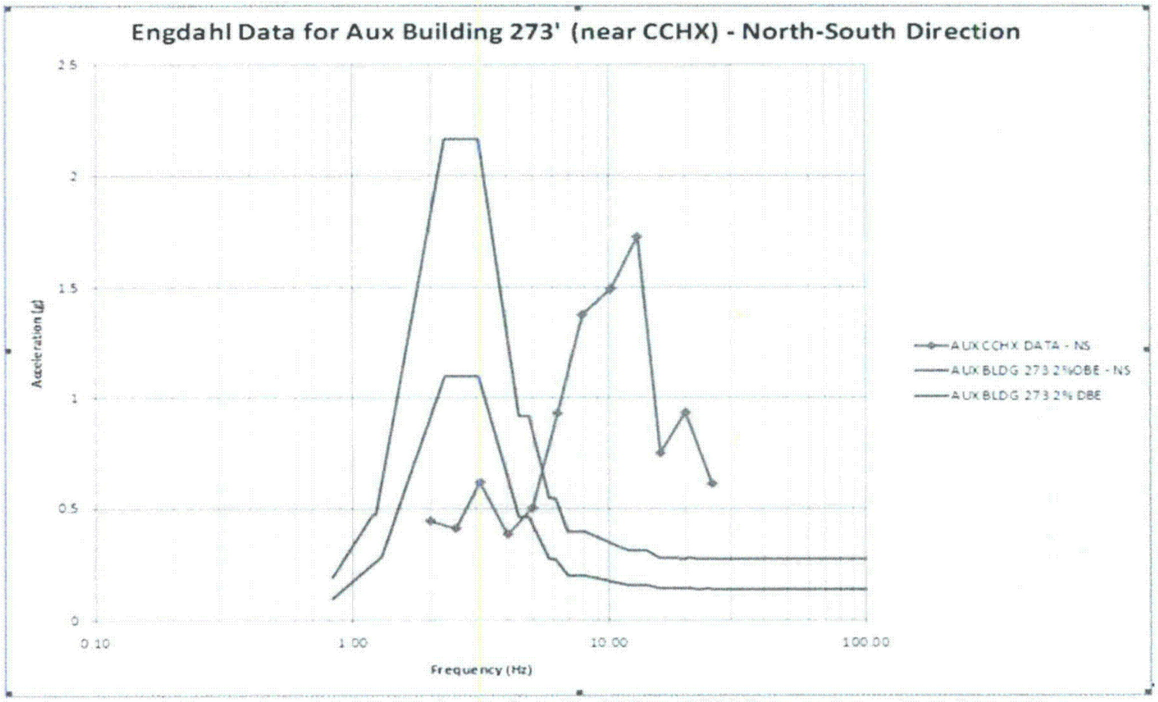


Figure 2.9 – Engdahl Data from Aux Building 273' Recorder – N-S Direction

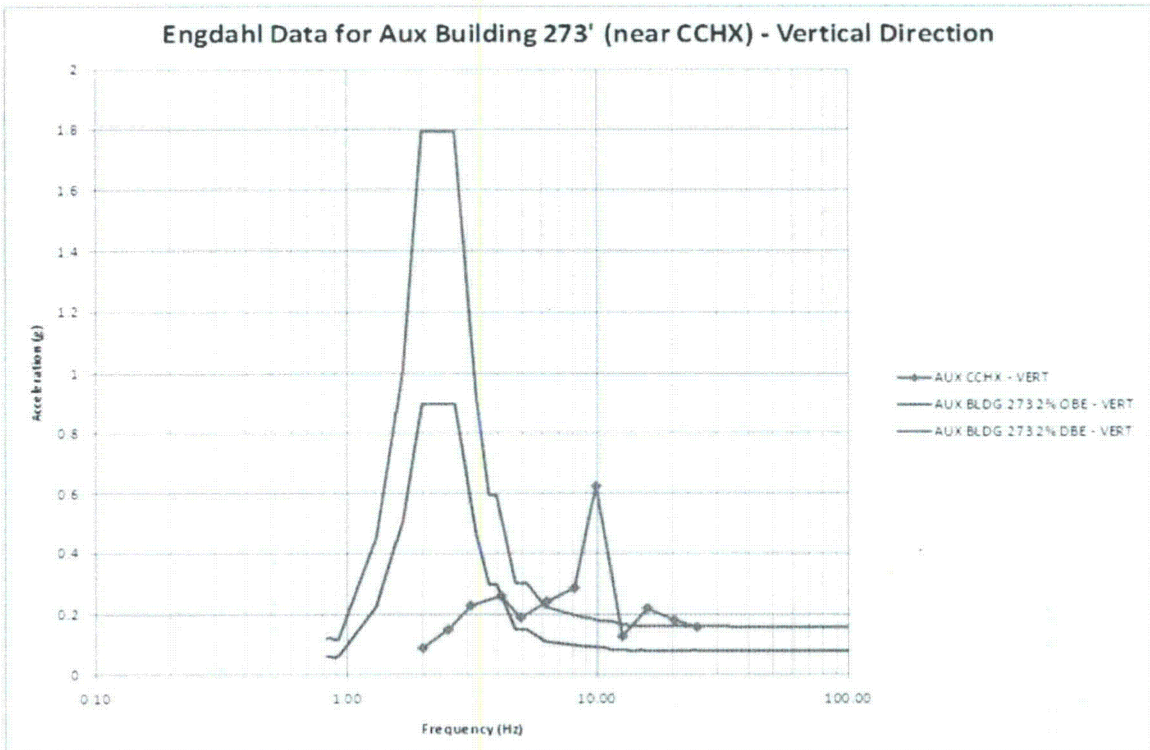


Figure 2.10 – Engdahl Data from Aux Building 273' Recorder – Vertical Direction

Kinemetrics Triaxial Recorder Data

Review and Comparison of the Data

The recorded data from the Kinemetrics Model FBA-3 Triaxial Time History Accelerograph Recorders were sent to the vendor for reading, calibration, and baseline correction. The results from the vendor are baseline-corrected time histories for the two elevations in Containment where these instruments are installed. The baseline correction appears to be reasonable based on a review of the velocity and displacement plots. The two elevations are the top of base mat elevation (216') and the 291' elevation. The time histories, which are digitized with 5 milliseconds intervals, are converted into response spectra up to 50 Hz. It is noted that the time interval of 0.005 second would give reasonably accurate spectral values up to about (1/.005)/10) or 20 HZ, and some approximations may be introduced up to about 50 HZ. However, this is judged to be reasonable. The response spectra created from the Kinemetrics data in the following pages was developed by Kinemetrics, and by Dominion engineering using finite element analysis software from the recorded, baseline corrected time histories provided from Kinemetrics. Both Bechtel and SGH also independently calculated response spectra using the same time histories. As expected, the spectra are the same from these sources. The data have been confirmed to be consistent.

Containment Basemat, Elevation 216'

For the Containment basemat elevation, the results are plotted at 5% damping against the design OBE and DBE response spectra corresponding to the top of the basemat. Further, a comparison is made to the IPEEE response spectrum created using the NUREG/CR-0098 median-centered site ground motion for rock as input. As noted previously, the design basis OBE and DBE spectra are exceeded by the recorded data in several frequencies; however, the recorded data is enveloped by the IPEEE response spectra curve.

See Figures 3.1 and 3.2 for horizontal and vertical comparison, respectively.

Average Exceedance

As shown in Figures 3.1 and 3.2, the data shows that the OBE and DBE were exceeded in all three directions; however, for purposes of quantifying how much the recorded response spectra exceeds the design basis earthquake spectra, the average exceedance is calculated in the frequency range from 2 to 10 Hz. This calculation is made for the north-south and vertical directions only since the recorded spectrum in the east-west direction is completely enveloped in that range by the DBE spectrum. The range of 2 to 10 Hz is chosen since this is the range of frequencies associated with damage to engineered structures and much of the plant equipment as defined in EPRI report NP-5930 and Reg. Guide 1.166.

Kinematics Triaxial Recorder Data

Table 3.1 – Average Exceedance from 2 to 10 Hz

VERTICAL DIRECTION				
Frequency [Hz]	DBE [g]	Recorded * [g]	Exceedance [g]	%Exceed
2	0.219	0.1318	No Exceed	-39.81%
2.25	0.219	0.1411	No Exceed	-35.59%
2.5	0.221	0.1686	No Exceed	-23.70%
2.75	0.222	0.2091	No Exceed	-5.79%
3	0.222	0.2386	0.0166	7.46%
3.25	0.222	0.2582	0.0362	16.30%
3.5	0.221	0.2630	0.0420	19.02%
3.75	0.22	0.2968	0.0768	34.92%
4	0.221	0.3097	0.0887	40.12%
4.25	0.221	0.2799	0.0589	26.63%
4.5	0.221	0.2756	0.0546	24.71%
4.75	0.224	0.2847	0.0607	27.12%
5	0.224	0.2823	0.0583	26.02%
5.25	0.227	0.2947	0.0677	29.81%
5.5	0.227	0.2861	0.0591	26.04%
5.75	0.227	0.2641	0.0371	16.33%
6	0.227	0.2573	0.0303	13.33%
6.25	0.227	0.2379	0.0109	4.82%
6.5	0.227	0.2535	0.0265	11.66%
6.75	0.227	0.2877	0.0607	26.75%
7	0.226	0.3191	0.0931	41.21%
7.25	0.224	0.3450	0.1210	54.03%
7.5	0.224	0.3649	0.1409	62.90%
7.75	0.224	0.3534	0.1294	57.77%
8	0.224	0.3379	0.1139	50.84%
8.25	0.226	0.3148	0.0888	39.28%
8.5	0.226	0.2993	0.0733	32.44%
8.75	0.226	0.2827	0.0567	25.10%
9	0.226	0.2621	0.0361	15.99%
9.25	0.226	0.2438	0.0178	7.89%
9.5	0.226	0.2551	0.0291	12.88%
9.75	0.226	0.2749	0.0489	21.65%
10	0.226	0.2911	0.0651	28.80%

AVERAGE EXCEEDANCE FROM 2 - 10 Hz	**21.12%**

NORTH-SOUTH DIRECTION				
Frequency [Hz]	DBE [g]	Recorded * [g]	Exceedance [g]	%Exceed
2	0.352	0.33836	No Exceed	-3.88%
2.25	0.359	0.35321	No Exceed	-1.61%
2.5	0.359	0.37645	0.0175	4.86%
2.75	0.361	0.40261	0.0416	11.53%
3	0.362	0.41980	0.0578	15.97%
3.25	0.362	0.43222	0.0702	19.40%
3.5	0.362	0.46852	0.1065	29.42%
3.75	0.361	0.45424	0.0932	25.83%
4	0.363	0.40768	0.0447	12.31%
4.25	0.363	0.41682	0.0538	14.83%
4.5	0.364	0.41936	0.0554	15.21%
4.75	0.366	0.41089	0.0449	12.27%
5	0.366	0.40077	0.0348	9.50%
5.25	0.366	0.42666	0.0607	16.57%
5.5	0.366	0.46472	0.0987	26.97%
5.75	0.366	0.46155	0.0955	26.11%
6	0.366	0.42878	0.0628	17.15%
6.25	0.366	0.40593	0.0399	10.91%
6.5	0.372	0.41769	0.0457	12.28%
6.75	0.372	0.41214	0.0401	10.79%
7	0.372	0.39544	0.0234	6.30%
7.25	0.372	0.38257	0.0106	2.84%
7.5	0.372	0.36939	No Exceed	-0.70%
7.75	0.372	0.36049	No Exceed	-3.09%
8	0.372	0.35955	No Exceed	-3.35%
8.25	0.372	0.36656	No Exceed	-1.46%
8.5	0.367	0.38500	0.0180	4.90%
8.75	0.367	0.40535	0.0383	10.45%
9	0.367	0.42293	0.0559	15.24%
9.25	0.367	0.43923	0.0722	19.68%
9.5	0.367	0.44551	0.0785	21.39%
9.75	0.367	0.44426	0.0773	21.05%
10	0.367	0.44791	0.0809	22.05%

AVERAGE EXCEEDANCE FROM 2 - 10 Hz	**12.17%**

Read directly from Kinemetrics Data Report ("NAPS Containment Mat 5% OBE SSE (DBE).PDF")

As shown above, the earthquake exceeded the DBE, on average, by about 12% in the N-S direction with the sharpest peak exceeding the corresponding DBE spectral ordinate by a factor of about 1.3, and about 21% in the vertical direction with the sharpest peak exceeding the DBE by a factor of about 1.6.

Kinematics Triaxial Recorder Data

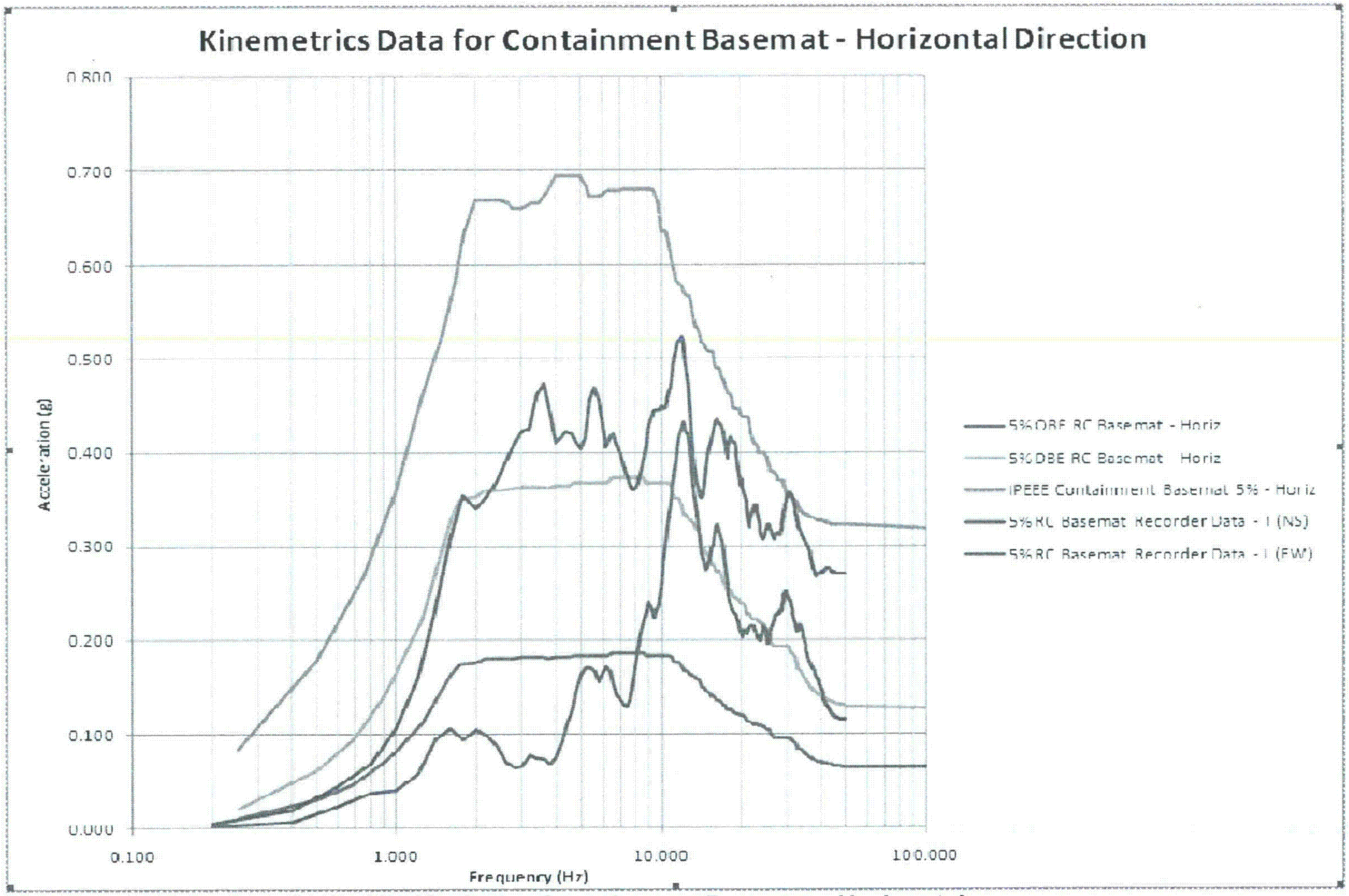


Figure 3.1 – Containment Basemat - Horizontal

Kinemetrics Triaxial Recorder Data

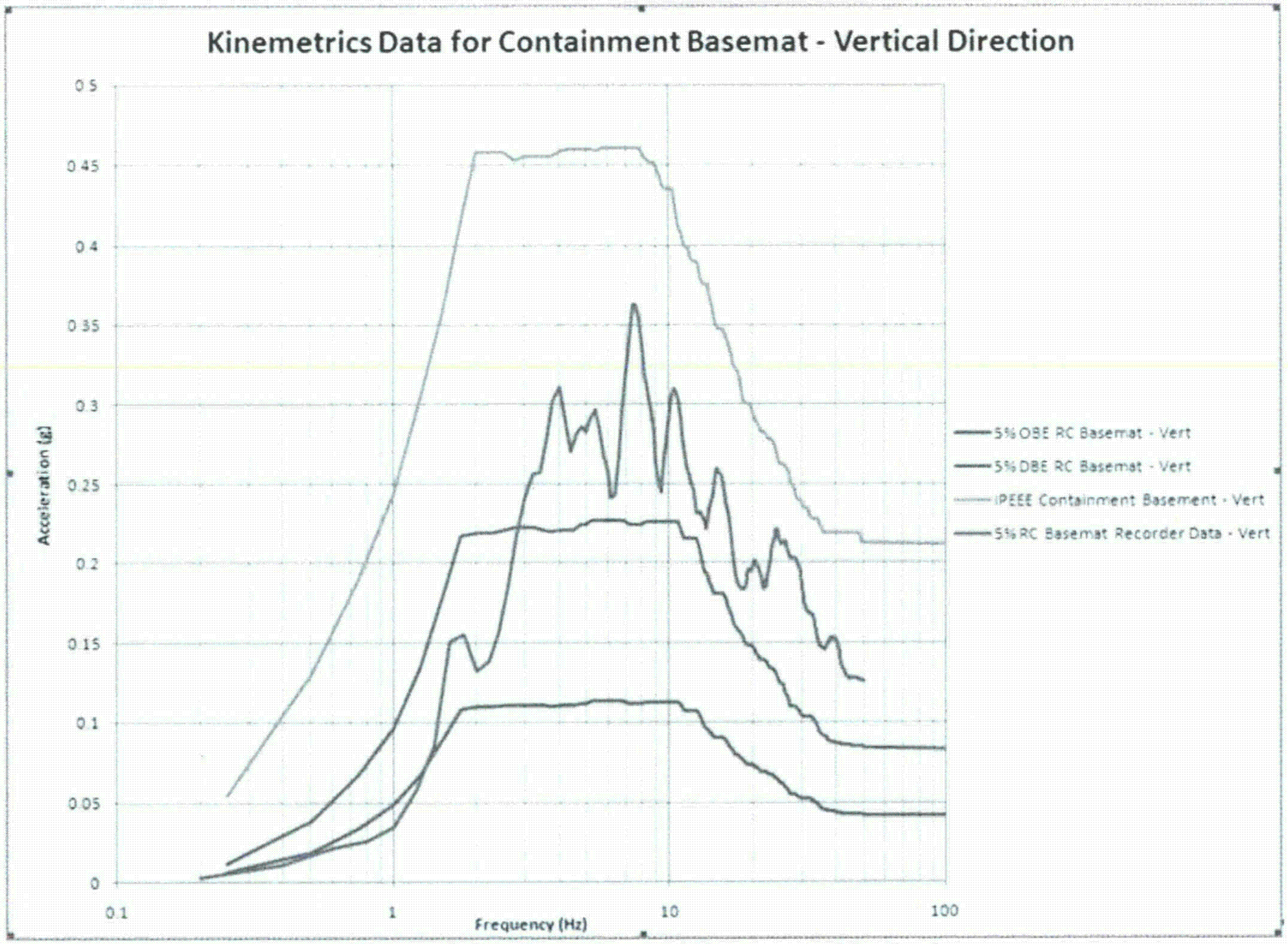


Figure 3.2 – Containment Basemat - Vertical

Kinemetrics Triaxial Recorder Data

Containment Operating Deck, Elevation 291'

For the 291' elevation of Containment, the Kinemetrics results are plotted at 5% damping against the design DBE response spectrum at the same elevation.

See Figures 3.3 and 3.4 for horizontal and vertical comparison, respectively.

Kinemetrics Triaxial Recorder Data

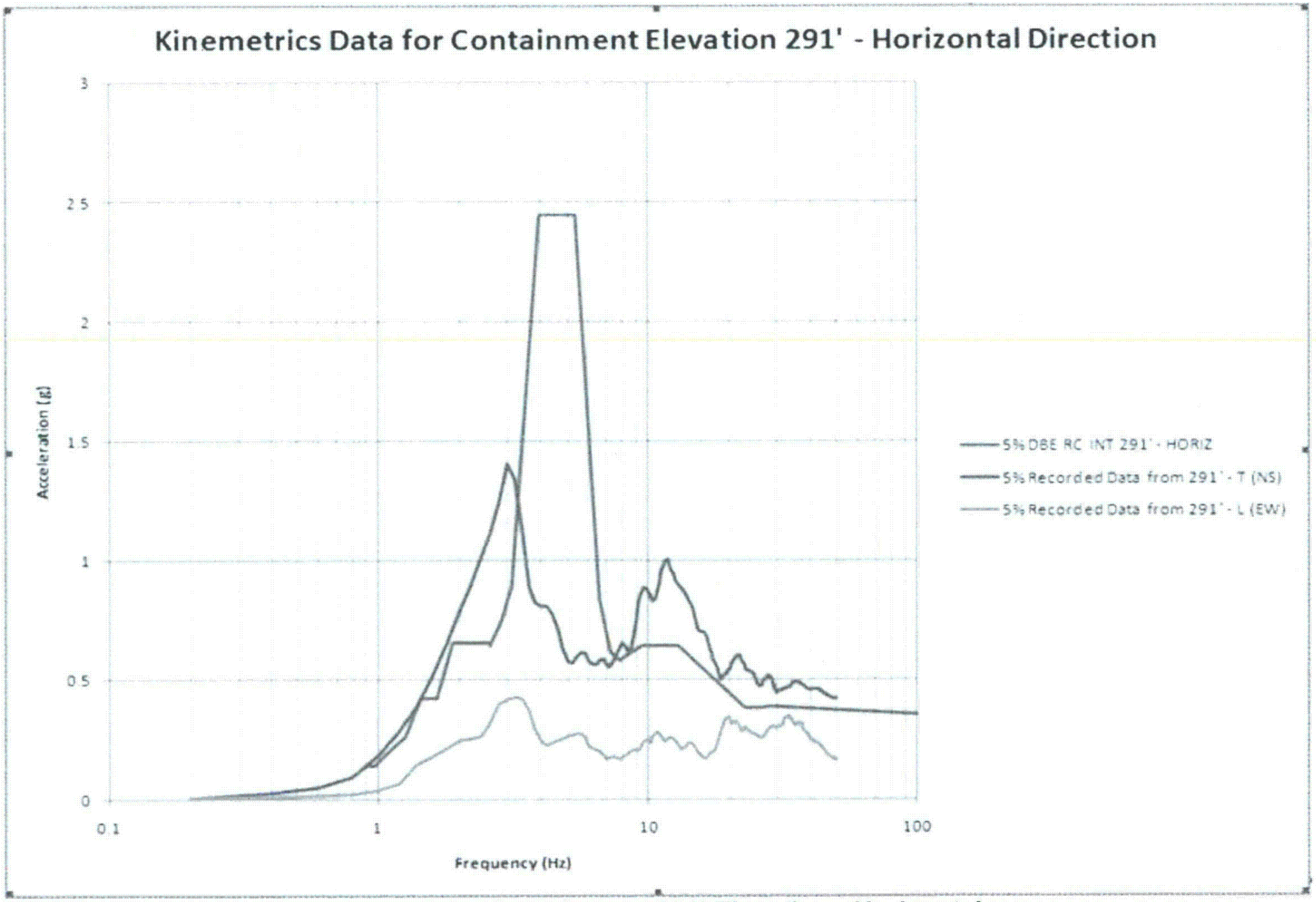


Figure 3.3 – Containment 291' Elevation - Horizontal

Kinemetrics Triaxial Recorder Data

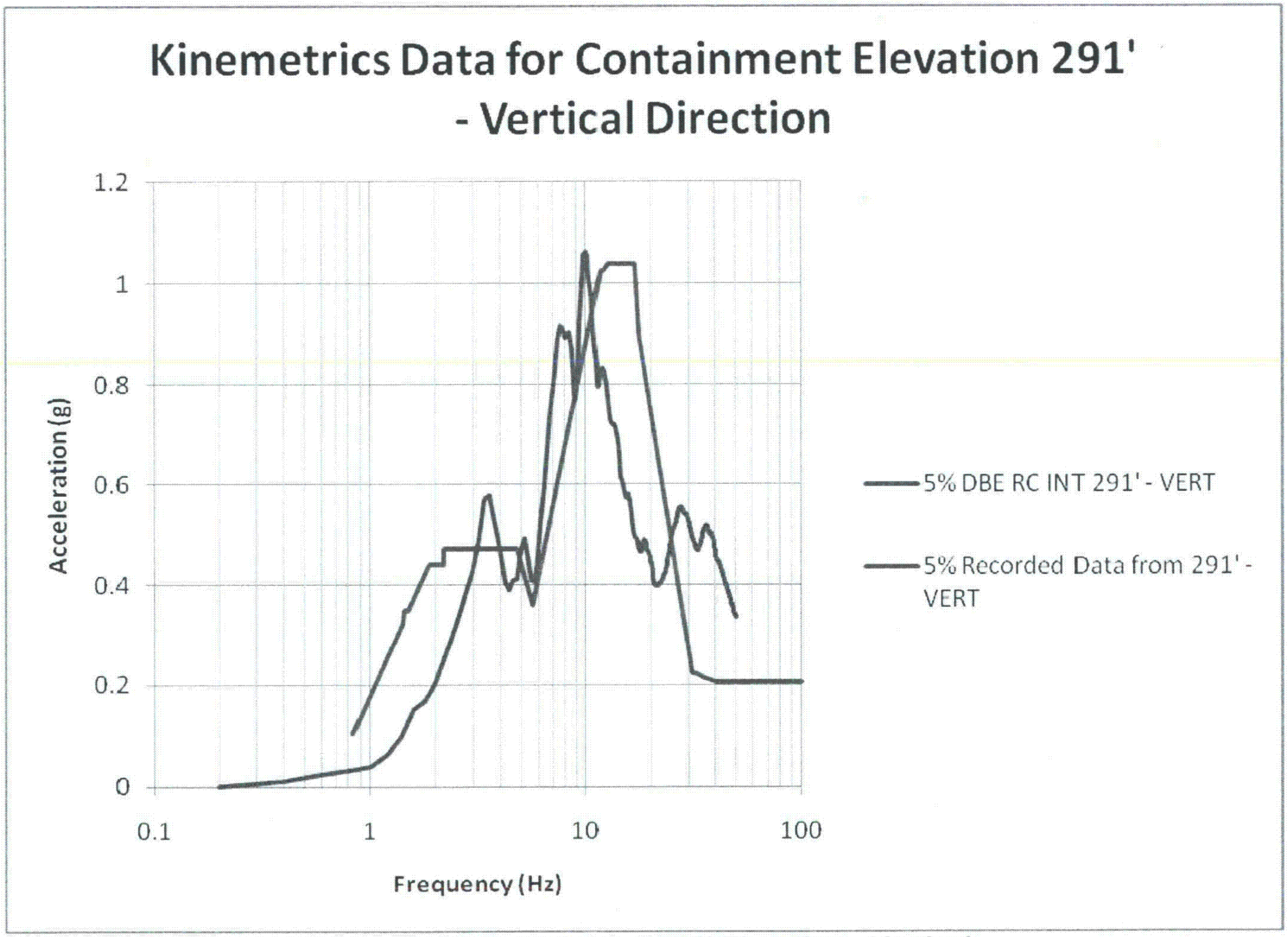


Figure 3.4 – Containment 291' Elevation - Vertical

Report of Inspections by a Seismic Review Team Including Industry Experts

Inspection Report Summary

On Friday, September 2, 2011, Dominion engineers were accompanied by several nuclear industry seismic experts for an inspection of North Anna Power Station looking for significant physical or functional earthquake-induced damage. The experts included the following individuals:

1. Robert P. Kennedy, Ph.D., RPK Structural Mechanics Consulting
2. Gregory S. Hardy, Senior Principal, Simpson, Gumpertz, & Heger (SGH)
3. Sanj Malushte, Ph.D., Bechtel Power Corporation
4. James R Martin Ph.D., Department of Civil Engineering and Environmental Engineering, Virginia Polytechnic Institute.
5. Russell A Green Ph.D., Department of Civil Engineering and Environmental Engineering, Virginia Polytechnic Institute.
6. Matthew R Eatherton Ph.D., Department of Civil Engineering and Environmental Engineering, Virginia Polytechnic Institute.
7. Martin C Chapman Ph.D., Department of Geosciences, Virginia Polytechnic Institute

The inspections were conducted in several plant locations and included a variety of plant equipment with an emphasis on areas/equipment where significant physical or functional earthquake induced damage would have been likely. Also, the inspection included non-safety related equipment and several of the low capacity items (i.e., HCLPF below 0.3g) identified during the IPEEE review. Areas and equipment inspected included the following:

- Emergency Condensate Storage Tanks (1-CN-TK-1) (On Grade - Yard)
- Refueling Water Chemical Addition Tank (1-QS-TK-2) (On Grade - Yard)
- Refueling Water Storage Tank (1-QS-TK-1) (On Grade - Yard)
- Primary Grade Storage Tanks (1-PG-TK-1A/B) (On Grade - Yard)
- Auxiliary Boiler Building (On Grade - Yard)
- Unit 2 Mechanical Equipment Room (291' elevation)
- Unit 2 Emergency Switchgear Room (254' elevation)
- Unit 2 Chiller Room (254' elevation)

Report of Inspections by a Seismic Review Team Including Industry Experts

- Unit 2 Main Control Room Area (273' elevation)
- Turbine Building (Multiple elevations), and
- Normal Condensate Storage Tanks (On Grade – Yard)

Inspection Results Summary

No significant physical or functional earthquake-induced damage was identified during the inspections conducted by the seismic expert review team. Some observations that could be speculated to be earthquake-induced damage were dispositioned as previously existing or not significant. A few minor non-earthquake related deficiencies were identified and are being tracked for disposition.

Enclosure 2

Post-Earthquake Inspections of Plant Structures, Systems and Components

Virginia Electric and Power Company
(Dominion)
North Anna Power Station

Post-Earthquake Inspections of Plant Structures, Systems and Components

EPRI-NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," was used to develop the North Anna methodology for performing inspections to assess significant physical or functional earthquake-related damage to structures, systems, and components (SSCs). EPRI-NP-6695 provides guidelines for responding to an earthquake that include immediate, post-shutdown, and long-term actions, and based on the findings during each stage, the level of additional effort required to ensure the plant is ready for operation. The following excerpt was taken from EPRI-NP-6695, Section 3.2, and details the recommended post-shutdown actions:

1. Focused Inspections. These are detailed, visual inspections of a pre-selected sample of representative structures and equipment. The equipment and structures included in the focused inspections should be selected to sample all types of safety-related equipment and structures found in the nuclear power plant, and should include equipment and types of structures which are considered most likely to be damaged from an earthquake. The focused inspections should also include non-safety-related, non-seismic equipment and structures which experience has shown to be of low seismic capacity to serve as earthquake damage indicators. These inspections should be performed by engineers experienced in the observation and evaluation of earthquake related damage to industrial and power facilities. The purpose of these inspections is to determine the need for expanded inspections and tests and to provide data to establish the EPRI Damage Intensity.

2. Determination of EPRI Damage Intensity. Using the information collected during the focused inspections and other observations, a group of experienced engineers should establish the EPRI Damage Intensity for the event using the guidelines presented in Section 2 of this report. Using the results of these inspections and the assessment of the EPRI Damage Intensity, the future course of actions needed to restart the plant are identified in the Figure 3-2.

3. Expanded Inspections. In the event that damage to the pre-selected sample of equipment or structures is found, or the EPRI Damage Intensity is determined to be 1 or greater, expanded inspections by qualified engineers should be undertaken to further define and evaluate potential damage to all components, systems and structures required for operation. This information can then be used to: (1) establish corrective actions and repairs that may be required to return the plant to a state of operational readiness, and (2) identify the need and timing for additional analytical and other engineering evaluations which may be prudent to assure the long-term integrity and reliability of the plant.

4. Surveillance Tests. Surveillance tests required by Technical Specifications should be performed to verify the operability of equipment needed for plant operation.

Initial visual inspections were performed by engineering personnel immediately following the August 23, 2011 earthquake, and the subsequent aftershocks up to

August 26, 2011. The damage discovered during these inspections did not identify any significant physical or functional damage to safety-related SSCs and only limited damage to non-safety related, non-seismically designed SSCs. Condition Reports (CRs) were submitted for the identified discrepancies. The results of these and additional focused inspections supported an EPRI Damage Intensity of 0, which is defined in Table 2-1 in EPRI NP-6695. To confirm the EPRI Damage Intensity, conservative measures were taken to perform comprehensive and methodical expanded inspections of the plant to further assess the impact of the earthquake on plant SSCs. The expanded inspections performed as part of the post-shutdown actions defined in EPRI NP-6695 are discussed below. Surveillance tests will also be completed prior to Unit 1 and 2 startups, respectively, to further demonstrate that SSCs can perform their design functions. The testing effort is also discussed in greater detail below.

System Inspections

The comprehensive system inspections included over eighty-systems for Unit 1 (which includes common systems) and over fifty-systems for Unit 2. These inspections were performed in accordance with station procedure 0-GEP-30, "Post Seismic Event System Engineering Walkdown," which was developed using the guidance provided in EPRI NP-6695. Inspection results were documented in procedure inspection logs, and discrepancies were entered into the Corrective Action System. The inspections were performed by qualified engineering personnel who had been trained on identifying seismic related damage.

Structural Component Inspections

The structural component inspections consisted of safety related and non-safety related structural components that meet regulatory requirements for Maintenance Rule and contribute to the operation of the station. These components are identified in procedure ER-NA-INS-104, "Monitoring of Structures North Anna Power Station," and the inspections were performed in accordance with this procedure. Attachment 8 of ER-NA-INS-104, "Seismic Event Inspection," was created based on the EPRI-NP-6695 guidelines and details the inspections to be performed on concrete structures, steel structures, and low pressure tanks. The inspection team looked for significant physical or functional damage caused by the earthquake that exceeded the acceptance criteria. The acceptance criteria are defined in procedure ER-NA-INS-104 and meet the guidelines established in EPRI NP-6695. The inspection results were documented in accordance with procedural requirements. The inspections were performed by qualified engineering personnel as defined in ER-NA-INS-104.

Detailed Inspections

In accordance with methods developed for the IPEEE, the plant was evaluated to a median-centered ground response spectrum shape anchored to 0.3g peak ground accelerations. Calculations were performed to determine the high-confidence-of-low-probability-of-failure (HCLPF) capacities of equipment and structures, and only a limited number of structures and components were determined to have HCLPF capacities below 0.3g. Thorough inspections of these components (with the exception of one group where relay chatter controlled the HCLPF capacity) are being performed by engineering personnel trained under the EPRI Seismic Qualification Utility Group (SQUG) course. The inspections are in progress and have not identified any evidence of significant physical or functional seismically related damage to date.

Electrical Inspections

For the 4160VAC, 480VAC, Vital/Semi-Vital 120VAC, and 125VDC equipment, the areas of focus consisted of four systems: Emergency Electrical (EE), Vital Bus (VB), Battery (BY), and Electric Power (EP). Comprehensive external inspections were performed in accordance with station procedure 0-GEP-30, "Post Seismic Event System Engineering Walkdown." Attachment 1, "Post Seismic Event Walkdown Checklist," of 0-GEP-30 contains the focus areas of these inspections for each type of equipment. In addition to the external inspections, an internal inspection was performed on the above mentioned equipment. This inspection was divided into categories of safety related systems and non-safety related systems. Safety related systems received nearly 100% internal inspections. For the non-safety related systems (EP), a sample of 10-15% of electrical cubicles, which contained various types of breakers and are located in several different plant locations and elevations, were internally inspected. Focus areas of the internal inspections were as follows: 1) Wiring pull-out from terminal blocks, 2) Damaged insulators (porcelain, ceramic, or plastic), 3) Wiring pull-out from lugs, 4) Wiring harness spacing issues, 5) Backed out or missing hardware from electrical bus work, 6) Foreign material, 7) Components that have become loose from electrical sockets, 8) Insulator damage to conductors, 9) Signs of electrical flashover, 10) Odd smells or sounds of resonance, and 11) Mechanical and Electrical misalignment. The inspection results were documented in the applicable inspection logs included in the procedure, and identified discrepancies were entered into the Corrective Action System.

A best effort visual inspection of the switchyard was initially conducted following the event. Additional inspections of the North Anna switchyard are currently being planned.

Potential for Hidden Damage

Although visual inspections have been performed, the possibility of "hidden damage", i.e., damage to SSCs that cannot be identified visually, was also considered. Based on the lack of significant physical or functional damage to safety-related plant SSCs, and only limited damage to non-safety related, non-seismically designed SSCs, identified in the System and Structural Component inspections, no hidden damage is expected. This is also based on review of industry insights from EPRI research related to the effect of the Niigata Chuetsu-Oki (NCO) earthquake of 2007 on the Kashiwazaki-Kariwa Nuclear Power Station (K-K) in Japan. Specifically, the NCO earthquake of 2007 was a Magnitude 6.6 earthquake that occurred on July 16, 2007 in the northwest Niigata region of Japan. The K-K plant is located approximately 15 miles from the epicenter of the earthquake. While the NCO earthquake exceeded the seismic design basis of the plant, it consisted of less than ten cycles of significant motion at frequencies generally less than 4Hz. EPRI's post-NCO earthquake peer review and plant walkdown inspection observed no significant damage to safety related SSCs but did observe consequential damage to non-safety related facilities, such as that resulting from soil collapse. As a result of the earthquake and the consequent damage to non-safety related SSCs, an extensive review of the SSCs in the K-K plant was completed.

EPRI established an expert panel to address the potential for hidden damage in SSCs that were subjected to the July 16, 2007 NCO earthquake near the K-K plant. The panel adopted a multi-element working approach addressing both experimental and analytical elements. The experimental elements included both seismic testing and earthquake field observations. The analytical elements included both structural analysis and physics of failure modes. Using the multi-element approach, the panel determined six specific equipment items and four general issues that had a potential for hidden damage. The plant owner, Tokyo Electric Power Company (TEPCO), investigated the ten items identified as significant by the EPRI panel, and in addition, investigated another fifteen areas which the EPRI panel had identified as less significant items. In total there were twenty-five potential non-visible damage items investigated in more detail by TEPCO. TEPCO reported to EPRI that they did not have any abnormal findings for the ten items identified as significant by the panel. A second panel also reviewed the potential for hidden damage in concrete structures. This panel concluded that if no surface cracks were visible, then there would be no hidden damage since non-surface opening (interior cracks) will not occur, and, even if small interior cracks did exist, the concrete was designed to accommodate the cracks.

Also, the two earthquakes, NCO and Mineral, Virginia, have several differences. Most notably is the fact that the K-K plant had significant damage to non-safety related SSCs, while North Anna had only limited damage to non-safety related SSCs. This is expected, as indicated by the CAV calculated from the ground motion recordings of the two earthquakes. The CAV values for the NCO earthquake are, on average, about six times higher than the CAV values for the Mineral, Virginia earthquake. Based on the review of the hidden damage evaluation process at the K-K plant and the lack of any abnormal findings, we believe that North Anna's comprehensive walkdown and inspection effort, together with functional tests of systems and components that are being completed after the August 23, 2011 seismic event, are adequate to conclude that North Anna SSCs do not have any further potential for hidden damage and that hidden damage in concrete structures is highly unlikely.

Nevertheless, additional activities are being pursued to further validate the expectation of no hidden damage. Vibration monitoring, pump/motor oil analysis, and thermography are being used during performance testing where applicable. Increased station awareness of operations, maintenance, and engineering personnel has resulted in identification of potential seismically induced concerns, although to date none of these concerns for SSCs has identified any significant physical or functional damage as a result of the earthquake. Extent of condition reviews as part of the normal corrective action process will also address any self-revealing "hidden damage". Also, in addition to the normal in-service inspection nondestructive examination activities planned for the ongoing Unit 2 refueling outage, additional Unit 1 and 2 sample weld inspections are planned for piping and pipe supports judged to be susceptible to a seismic event.

Although no hidden damage is expected based on the minimal damage observed visually, the activities discussed above are deemed adequate to identify any hidden damage from the seismic event, should it exist. In addition, the following specific actions have been/are being taken: 1) buried piping system pressure tests are being performed on the buried portions of Quench Spray, Recirculation Spray, and Service Water System piping, 2) dry transformers were inspected as part of the electrical system inspections, 3) station batteries were inspected with thermography, as well as individual cells checked using the associated periodic test, 4) float switches and induction relays are being tested as part of the scheduled functional tests, 5) long vertical pumps (e.g. Low Head Safety Injection and Outside Recirculation Spray Pumps) were tested, and vibration and tribological results were acceptable, 6) expansion anchor bolts were inspected, and some tightness checks were performed, and 7) a sampling of electrical connections were tested for tightness, and found to be acceptable.

Inspections of the North Anna Service Water Reservoir and the Main Dam

The North Anna Power Station Service Water Reservoir and Main Dam structures were inspected and evaluated following the August 23, 2011 seismic event. The Service Water Reservoir structure is classified as safety-related since it provides cooling for the Recirculation Spray Heat Exchangers when they are called upon to provide containment heat removal during a loss of coolant accident. The Main Dam is categorized as non-safety related with regulatory significance (NSQ) as it impounds Lake Anna and provides circulation water to the plant.

Available piezometric and settlement data at the Service Water Reservoir, and piezometric and drainage weir data at the Main Dam have been examined in response to the seismic event. This information, together with visual inspection observations, has been analyzed to determine if these structures and their appurtenances remain stable and capable of performing their design functions. The instrument data indicates that the pore water pressures and deformations are generally within the expected fluctuations, or are at levels that are insignificant and within typical structural tolerances for the facilities being considered. Inspections of these structures corroborated well with the instrument data and did not indicate any issues that would compromise their design functions. Based upon the available instrument data and the inspection observations, the Service Water Reservoir and the Main Dam sustained no significant physical or functional damage and remain capable of performing their intended design functions.

Inspection Results Summary

Comprehensive and methodical inspections of North Anna Units 1 and 2 SSCs were completed in accordance with station procedures. These procedures were created

/revised to incorporate EPRI NP-6695 guidance regarding post-shutdown inspections following a seismic event. The expanded inspections did not identify any significant physical or functional damage to safety related SSCs that would render them incapable of performing their design functions. Reported damage and observations from the earthquake include spurious actuation of the sudden pressure relays on the Reserve Station Service Transformers, limited cracking of ceramic / porcelain components on switchyard equipment, limited cracking of non-safety related walls, and movement of the Independent Spent Fuel Storage Installation casks. The most significant visual damage of a non-safety related SSC was spalled concrete on a condensate polishing tank support pedestal that did not affect function. The most significant damage that required repair of non-safety related equipment was Generator Step Up (GSU) transformer bushing leakage. This is a heavy bushing that projects out from the transformer and is primarily supported on one end; thus, this damage is not surprising. Other than the above items, no significant physical or functional seismically induced damage to non-safety related SSCs has been observed. The inspection results support an EPRI Damage Intensity of 0. Furthermore, based on the inspection results, as well as the EPRI research related to the effect of the 2007 NCO earthquake on the K-K nuclear plant, hidden damage is not expected to have occurred.

Surveillance Tests

Section 5 of EPRI NP-6695 provides guidelines for post-shutdown inspections and tests of nuclear plant equipment and structures required for operation prior to restart of a nuclear plant which has been shut down due to an earthquake which exceeds the OBE. To further evaluate the effect of the earthquake on the functionality of nuclear plant equipment, it recommends that surveillance tests, required to verify that the limiting conditions for operation as defined in the plant Technical Specifications (TS) are met, also be performed.

A Unit 1 and a Unit 2 list of the surveillance tests to be performed have been developed using guidance from EPRI NP-6695, Appendix B, "Typical Surveillance Tests for PWRs." To ensure a comprehensive test program is completed prior to restart, additional testing has also been included. Surveillance tests are being performed to demonstrate the availability and operability of components and systems important to nuclear safety or required to mitigate the consequences of an accident as identified in the TS. This will result in well over 400 surveillance tests being performed for Unit 1 prior to restart. In addition to the tests normally performed during a refueling outage, over 150 additional surveillance tests are being performed for Unit 2 prior to its restart.

Enclosure 3

Post-Earthquake Evaluation of the Reactor Vessel Internals

Virginia Electric and Power Company
(Dominion)
North Anna Power Station

Post-Earthquake Evaluation of the Reactor Vessel Internals

The impact of the August 23, 2011 seismic event in Virginia on the ability of North Anna Units 1 and 2 Nuclear Steam Supply System (NSSS) Reactor Vessel (RV) internals to perform their design bases functions was assessed and is summarized below.

Design basis functions of the reactor internals are given in the North Anna Updated Final Safety Analysis Report (UFSAR) Section 4.2.2.

The design functions of the RV internals can be met as long as their structural integrity has been maintained. For this evaluation, the criterion used for structural integrity is that no dimensional changes occur (i.e., there is no yield in internals components). If this criterion is met, dimensions of the RV internals components are maintained. The RV internals will thus continue to perform their design functions. In this evaluation, the structural integrity confirmation is assessed by application of a conservative criterion to calculated load results from either the Operating Basis Earthquake (OBE) or Design Basis Earthquake (DBE), to confirm that margins exist.

Basis for Conclusion of Functionality

As noted in Enclosure 2, evidence of inspections is consistent with Damage Intensity 0 on the EPRI seismic damage scale. EPRI NP-6695 describes how prescribed inspections and tests are keyed to the severity of the earthquake. No specific inspections of reactor internals or associated components are specified in EPRI NP-6695 for Intensity 0 earthquakes. Since the earthquake produced only minimal damage to non-seismically designed equipment, and since there was no significant physical or functional damage to seismically designed systems, structures, and components that were examined following the event, there is a reasonable assurance that there was no significant physical or functional damage to RV internals, and that the RV internals remain functional and capable of performing their design functions. Additional evaluations of RV internals design margins have been performed based on existing design analyses of the structural integrity of the RV internals. These evaluations, in addition to the above-described reasonable assurance of continued functionality, support the conclusion that the earthquake resulted in no significant physical or functional damage to the RV Internals, and that the RV internals remain capable of performing their design bases functions.

Evaluation of RV Internals Loadings

The details of the dynamic analyses, input forcing functions, and response loadings are presented in UFSAR Section 3.9.1.2.3. The RV internals are modeled to determine dynamic loads produced by a reactor coolant loop (RCL) branch line pipe rupture (for both cold-leg and hot-leg breaks), and for the response due to operational-basis and design-basis earthquakes.

The following events are considered in the structural analysis of the RV internals:

1. Loads produced by a RCL branch pipe rupture for both cases (LOCA): cold-leg and hot-leg break.
2. Response due to a DBE.
3. Maximum stresses obtained in each case are added in the most conservative manner.

Only the loads calculated for a seismic event (either the OBE or DBE) are of interest for this evaluation. Calculated results from existing design analyses were evaluated for several key RV internals interface load points in the vessel. The calculated seismic-only loads were compared with allowable load limits which correspond to allowed stress limits for Upset conditions (Normal + OBE Loads) for which no deformation is allowed. This provides a more stringent criterion than is typically applied to the DBE loads when assessed in normal design calculations (UFSAR 3.9.3.1.1). This conservative criterion provides additional assurance that RV dimensions and geometry are maintained. The interface loads evaluated satisfied this criterion.

This provides a reasonable assurance that even though the seismic event of August 23, 2011 exceeded the OBE and DBE the RV internals will continue to satisfy their design bases functions. This conclusion is based on the system-wide evaluation above, using conservatively calculated design loads. To augment this analytical-based evaluation, one long-term action is identified relating to reactor vessel internals. This action is to develop a plan with the Nuclear Steam Supply System (NSSS) vendor consisting of additional evaluations or inspections, as warranted, to assure long term reliability of the reactor internals for North Anna Unit 1 and 2 (Enclosure 9). In addition, visual examination of the RV internals will be conducted following the Unit 2 fuel offload. Any identified discrepancies would be appropriately dispositioned through the Corrective Action System.

Conclusions

Results of system inspections and walkdowns conducted following the August 23, 2011 seismic event are consistent with Intensity 0 on the EPRI seismic damage scale [EPRI NP-6695 and Enclosure 1]. No significant physical or functional damage to seismically designed components (which includes reactor internals) is expected for an Intensity 0 event. No specific inspections of reactor internals are recommended in EPRI NP-6695 for an Intensity 0 classification. Since the earthquake produced only minimal damage to non-seismically designed equipment, and since there was no significant physical or functional damage to seismically designed systems, structures, and components that were examined following the event, there is a reasonable assurance that there was no significant physical or functional damage to RV internals, and that the RV internals remain functional and capable of performing their design functions. RV internals are designed to withstand combined seismic and LOCA forces; calculated loads on RV internals are dominated by LOCA forces, which did not occur during this event. The North Anna RV internals have been evaluated for loads generated during a seismic event (either OBE or DBE), using results of existing design analyses. These conservatively calculated loads have margin when compared to a conservative allowable load (applicable to the OBE). These evaluations support the conclusion that the earthquake resulted in no significant physical or functional damage to the RV Internals, and that the RV internals remain capable of performing their design bases functions.

Enclosure 4

Post-Earthquake Assessment of New and Irradiated Fuel

Virginia Electric and Power Company
(Dominion)
North Anna Power Station

Post-Earthquake Assessment of New and Irradiated Fuel

An earthquake occurred in the vicinity of the North Anna Power Station on August 23, 2011. As a result of its potential impact on station equipment, verification of the acceptability of the fuel assemblies and non-fuel core components in the new fuel storage area, the spent fuel pool, and the Unit 1 and Unit 2 cores must be performed. The purpose of this review is to identify the inspections that have been or will be performed to confirm that the recent seismic event at North Anna did not result in significant physical or functional damage to the fuel assemblies and the fuel insert components. These inspections allow for confirmation of the condition of both the new and spent fuel, as well as non-fuel core components such as control rods and burnable poison assemblies.

Discussion

Dominion is working with AREVA, the current fuel supplier for North Anna, to assess the margins in the fuel. For this evaluation, the acceptance criterion is that no plastic deformation is predicted. In addition, Dominion – with AREVA's input - has compiled a list of inspections to be conducted for fuel and fuel inserts in the new fuel storage racks and spent fuel pool, and during offload of the Unit 2 core, to verify the acceptability of the Unit 2 fuel for use or reuse. Unit 2 fuel will be examined prior to the Unit 1 startup. The Unit 2 fuel will be used to assess the condition of the Unit 1 fuel. If the Unit 2 fuel meets all of the inspection criteria described herein, no inspections of Unit 1 fuel are planned.

The EPRI "Guidelines for Nuclear Plant Response to an Earthquake" (Reference 1) only mentions fuel and controls rods briefly. Results of physical inspections indicate the seismic event damage is consistent with Intensity 0 on the EPRI seismic damage scale. Reference 1 describes how prescribed inspections and tests are keyed to the severity of the earthquake. No specific inspections of fuel or associated components are specified in Reference 1 for Intensity 0 earthquakes. Since the earthquake did not produce any significant physical or functional damage to safety-related plant SSCs and only limited damage to non-safety related, non-seismically designed SSCs that were examined following the event, there is reasonable assurance that there was no significant physical or functional damage to the fuel, and that the fuel remains functional and capable of performing its design functions. The inspections described herein provide additional confirmation of the expectation that the earthquake resulted in no significant physical or functional damage to the fuel or fuel inserts, and that they remain fully functional and capable of performing their design basis functions.

Miscellaneous Inspections to Support Fuel Inspections

The following inspections have been performed:

- Prior to any movement of fuel assemblies for inspection, the handling equipment - including handling tools, new fuel elevator and bridge crane - was verified as operational using functional checkouts required in the fuel handling procedures.

- The racks are predicted to move during a DBE (Reference 2). The spent fuel storage rack arrays were inspected to confirm that the racks had not shifted significantly or become distorted during the earthquake. This was accomplished by verifying that the indexing used on bridge crane remains accurate and can still be used to remove or insert fuel assemblies into rack. The indexing coordinates were verified by inserting and removing the dummy fuel assembly in at least one empty spent fuel cell in each rack. Recent videos of the rack cells taken after the earthquake were also compared with previously existing videos of the racks to confirm there was no apparent damage of the supports for the storage racks.
- The dummy fuel assembly was lifted and visually inspected prior to its use for any other system checkouts or verification.

New Fuel Storage

At the time of the earthquake, there were eighteen new fuel assemblies in the new fuel storage area, eleven of which contained burnable poison rod assemblies (BPRAs). In addition, there was one new BPRA hanging from a support plate in a new fuel storage cell. The 18 fuel assemblies were free standing in their storage cells and thus able to move and contact the cell walls during a seismic event. There is slightly more than ½ inch clearance between the cell and the assembly if it is sitting to one side of the cell. These 18 assemblies were visually inspected for any evidence of impact between the storage cell and the grids or any other parts of the assembly. This inspection was performed when the assemblies were moved to the spent fuel pool and was more involved than the normal new fuel receipt inspections. AREVA provided recommendations on the scope and criteria to be used during these inspections. All 18 assemblies were found to satisfactorily meet the inspection criteria.

Prior to moving any assemblies, an inspection of the underneath portion of the New Fuel Storage area was conducted to ensure there was no significant physical or functional damage or distortion that would lead to interferences between the assemblies and the storage cells when raising the fuel assemblies. There were no issues identified from that inspection that indicated conditions exist that would result in any adverse impact on the fuel.

The eleven BPRAs that were in new fuel assemblies were each lifted a short distance by hand and lowered back into the assembly to ensure that they would self-seat. Additional inspections were performed on these eleven BPRAs in accordance with the AREVA recommendations, including inspections of the nuts and welds connecting the poison rodlets to the BPRA baseplates and, while the BPRAs were slightly raised, inspecting the BPRA rodlets for dents or abrasions to the extent possible. The BPRA that was currently hanging from the support plate was inspected when it was removed from the support plate and placed in a fuel assembly. AREVA provided separate inspection recommendations for this BPRA. The inspected BPRA were determined to satisfactorily meet the inspection criteria.

Spent Fuel Pool

The spent fuel pool rack cells are 8.875 inches square on the inside. There is slightly less clearance between the fuel and the cell walls in the spent fuel racks compared to the new fuel storage cells, and the potential for fuel damage in the spent fuel pool is further reduced by hydraulic damping effects. Nevertheless, the following inspections were performed:

- Five new fuel assemblies scheduled for use in Unit 2 Cycle 22 that were placed into the spent fuel pool prior to the earthquake were video inspected for any signs of damage. These assemblies were inspected in accordance with the recommendations provided by AREVA.

- During preparation of the spent fuel pool prior to the Unit 2 offload, a pre-offload fuel shuffle was performed. During this shuffle, a sample consisting of ten of these assemblies was also video inspected for any signs of damage.

When inspecting these irradiated assemblies, recommendations provided by AREVA were used to supplement Dominion's normal criteria for irradiated fuel inspections. The population of new fuel assemblies and pre-offload shuffle assemblies inspected provides a representative sample of the fuel designs and storage locations across the spent fuel pool. The fuel assemblies examined during these inspections satisfactorily met the inspection criteria.

Unit 1 and Unit 2 Cores

There are currently no failures in the Unit 2 core and an estimated two failed rods in the Unit 1 core, which were identified earlier in the operating cycle. The Unit 1 and Unit 2 RCS coolant activity following shutdown was consistent with the known fuel condition at the time of the earthquake, and indicated that no fuel failures occurred in either unit as a result of the earthquake.

The lateral clearances between fuel assemblies and between the fuel assemblies and the core baffle are very small. It is expected that any impact loading between assemblies was small enough that no damage to grids would have occurred. Binocular visual inspections of the Unit 2 assemblies are conducted during offload and during normal detailed visual inspection (using video) of a sample of the assemblies in the core. During these video inspections, additional attention will be given to the grids to look for distortion or any deflection of the inner grid straps and mixing vanes. As necessary, assembly movement will be stopped at grid elevations and camera angles will be varied to allow the best possible visual inspection of the grid structure. If the vertical acceleration was sufficient to lift the core and compress the top nozzle hold down springs, some indications may appear on the springs or on the corner pads if the springs bottomed out. Detailed inspections of the side of the nozzle when the benchmark inspections are being performed should identify any such damage to the nozzles.

Dominion fuel inspection procedures requires inspection of each fuel assembly with binoculars or by camera during the offload, and a detailed video inspection of a sampling of assemblies after the offload. These binocular and video inspections are normal outage scope work. During the Unit 2 fuel offload, the 157 assemblies in the core will be visually inspected using binoculars or a camera for any signs of damage. Prior to the earthquake, thirteen assemblies had been selected for inspection during the North Anna 2 refueling outage, and video inspections will also be performed on additional assemblies recommended by AREVA that resided in core locations that are most susceptible to seismic damage. This level of video inspection is consistent with the approach identified in Reference 3 (only inspecting a sample of the fuel assemblies since no fuel failures were indicated by the Unit 2 radiochemistry data). Additional inspection criteria provided by AREVA will be used during these detailed video inspections.

Based on recent North Anna experience, it is expected that the visual inspections during the Unit 2 core offload will identify excessive fuel rod bow in some of the fuel assemblies. Any of these assemblies that are slated for reuse will also be inspected for any sign of seismic damage during the detailed video inspection campaign, using the AREVA criteria. Fuel assemblies with excessive fuel rod bow that are not planned for reuse will be inspected to normal Dominion criteria.

The operability determinations (ODs) previously prepared for the fuel rod bow concern in AREVA fuel will be formally reviewed, although the recent seismic event is not expected to impact the fuel rod bow ODs. The RCS radiochemistry data indicates there are no fuel rod failures in Unit 2, and shows no indications of new fuel failures in either unit resulting from the earthquake. Additionally, grid deformation that might be seen during a combined seismic and LOCA event would not impact the fuel rod bow phenomenon. The fuel assembly's ability to maintain a coolable geometry is the main concern associated with grid deformation. While the fuel rod bow phenomenon causes the closure of one water channel, it also opens the water channel on the opposite side. Therefore, the assembly's ability to maintain a coolable geometry is not compromised by fuel rod bow, even with possible grid deformation.

Acceptable results of the visual inspections performed during the Unit 2 offload, together with acceptable results of the detailed video inspections of fuel assemblies in the core during the earthquake that are inspected to the additional criteria provided by AREVA, will support the continued use of Unit 1 fuel without inspection.

When the units tripped during the recent seismic event, all control rods fully inserted. However, testing will be performed to confirm that the rod cluster control assemblies (RCCAs) still freely travel within the fuel assembly guide tubes. After the Unit 2 offload, the RCCA drag loads will be measured in the spent fuel pool to assess whether the fuel assembly or the RCCAs have any distortion. Post-latch drag testing and hot rod drops of the RCCAs are already required as part of the normal start-up activities and will insure that the RCCAs and CRDMs are functional. A video inspection of the RCCA central hubs will be performed to provide additional confirmation of RCCA integrity. A satisfactory assessment of the Unit 2 RCCAs (rod drag measurements and spent fuel pool video inspections) will provide assurance that the Unit 1 RCCAs are in a similar condition. Although normally required only at BOC, hot rod drop testing of the Unit 1 RCCAs in accordance with normal station procedures will be performed prior to the restart of Unit 1 to confirm the continued acceptable condition of the Unit 1 RCCAs.

Conclusions

Table 1 lists the inspections that were mentioned in this Enclosure. Completed tasks are identified.

The inspections listed in this Table will permit assessment of the condition of the fuel assemblies in the new fuel storage area, spent fuel pool, and North Anna Unit 1 and Unit 2 cores at the time of the earthquake. If the inspection results are satisfactory, the fuel assemblies can be considered acceptable for use or reuse. If any significant seismic damage is observed during any fuel inspections, then the scope of the program (defined herein) will be reassessed.

References

1. EPRI Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.

2. "A Summary of Information in Support of Increasing the Spent Fuel Storage Capacity at North Anna Power Station Units 1 and 2," Attachment 3 to Letter from R. H. Leasburg to H. R. Denton (U. S. NRC), "Amendment to Operating Licenses NPF-4 and NPF-7, North Anna Power Station Unit Nos. 1 and 2, Proposed Technical Specification Changes," Serial Number 450, August 20, 1982.

3. EPRI Report No. 1016317, "EPRI Independent Peer Review of the TEPCO Seismic Walkdown and Evaluation of the Kashiwazaki-Kariwa Nuclear Power Plants," January 2008.

Table 1
Fuel and Miscellaneous Inspections

Area	Task	Status
New Fuel Storage	Inspect BP1729 (hanging from support plate in new fuel storage cell) when transferring to new fuel assembly 13L. Use AREVA Inspection criteria.	Complete
	Inspect 18 New Fuel Assemblies prior to transfer to SFP. Use AREVA inspection criteria.	Complete
	Inspect 11 BPRAs in Fuel Assemblies. Use AREVA inspection criteria.	Complete
	Drag test 7 New Fuel Assemblies.	Complete
Spent Fuel Pool	Inspect a sample (10) of assemblies during the pre-offload shuffle. Video inspect according to the normal benchmark video inspection requirements.	Complete
	Inspect a sample (5) of new fuel assemblies and BPRAs. Use AREVA inspection criteria.	Complete
Unit 1 Core	Hot Rod Drop Tests. Follow standard North Anna procedure used at BOC.	See Enclosure 8
Unit 2 Core	Prior to core offload, inspect top nozzle locking lug position of two assemblies. Ensure positive lock of the quick disconnect mechanisms.	Complete
	Verify RCCAs still freely travel within the fuel assembly guide tubes. Measure RCCA drag loads in the spent fuel pool.	See Enclosure 8
	Perform routine binocular visual inspection during core offload. Any anomalous conditions will be video inspected.	See Enclosure 8
	Perform video inspections on 13 benchmark assemblies and AREVA recommended fuel assemblies. The AREVA recommendations include fuel assemblies from specific core locations susceptible to grid damage during seismic events.	See Enclosure 8
	Perform video inspection of RCCA central hubs.	See Enclosure 8
	Perform video inspections on assemblies with any anomalies observed during binocular inspections. Part of normal outage scope. Normal Dominion irradiated fuel inspection criteria apply. Assemblies planned for reuse will also be inspected to the AREVA criteria.	See Enclosure 8

Area	Task	Status
Unit 2 Core (cont.)	Post-latch drag testing and Hot Rod Drops. Both are part of normal outage scope and will follow standard North Anna procedures.	See Enclosure 8
Miscellaneous Inspections	Prior to picking up any fuel, verify that all handling equipment including handling tools, new fuel elevator, and bridge crane are operational prior to fuel inspections.	Complete
	Visually inspect dummy fuel assembly prior to picking up other fuel in spent fuel pool and prior to moving new fuel to pool.	Complete
	Visually inspect spent fuel storage racks for indications of significant rack movement and distortion prior to fuel movement. Functionally verify no significant change to indexing coordinates by inserting and removing dummy fuel assembly at a minimum of one per rack.	Complete

Enclosure 5

Post-Earthquake Assessment of Spent Fuel Storage Racks

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station**

Post-Earthquake Assessment of Spent Fuel Storage Racks

The purpose of this assessment is to summarize the results of inspections and testing that were performed to confirm that the spent fuel storage racks are able to perform their intended design function after the seismic event on August 23, 2011. Inspections and functional testing of the spent fuel storage racks were performed to confirm their current structural condition prior to the offload of fuel assemblies from either core.

Spent fuel storage rack descriptions and design function from Reference 1 and consistent with UFSAR Section 9.1.2:

> *The spent fuel storage racks are classified seismic Category I and are designed to withstand the effects of the DBE and yet remain functional and maintain subcriticality.*
>
> *The storage cell structure, acting in concert with the rack base and the rack support feet, provides the structural strength and stiffness characteristics required for the rack to accommodate the applicable seismic accelerations presented for NAPS. No wall bracing or attachments are required to support the fuel racks under any design condition. Sufficient space is provided between adjacent spent fuel racks to preclude impact/collision in the event that sliding occurs during a seismic event.*

The spent fuel pool criticality analysis of record was submitted to the NRC and subsequently approved in References 3 and 4. This analysis eliminated the spent fuel storage rack Boraflex credit. Therefore, the post-seismic condition of the Boraflex panels is not important to the design function of the spent fuel storage racks.

The following inspection and test have been performed on the spent fuel storage racks post-earthquake:

1. Foreign Object Search and Retrieval (FOSAR) video of the spent fuel storage racks, and
2. Insertion of the dummy fuel assembly into two empty cells in all 16 spent fuel storage racks.

The FOSAR video inspection (performed on September 5, 2011) shows the entire spent fuel storage racks after the earthquake. Review of this video does not show any significant physical or functional damage to the spent fuel storage racks (i.e., the racks as a whole are still fully intact and supporting all the fuel assemblies as intended by the design). The racks have sufficient spacing, and there is no evidence that they came into contact with each other or the walls of the spent fuel pool. In particular, the spacing between rack modules and the gap between the rack modules and the wall appear unchanged from pre-earthquake conditions. Sliding of the racks during an earthquake is consistent with the design and hypothesized seismic behavior, as stated in the design function discussion above. The observation that the post-seismic spent fuel storage rack gaps are not abnormal and no significant physical or functional damage was found supports a conclusion that even if sliding occurred, the spent fuel storage racks are still properly positioned, and continue to be able to support the movement and storage of fuel assemblies.

To confirm the structural integrity of the spent fuel racks, a sample of images of individual spent fuel storage cells in the September 5, 2011 FOSAR video were viewed in detail, and compared to existing images from pre-earthquake fuel inspections in the spent fuel pool[1] to determine if any change in appearance occurred that could possibly be attributed to the earthquake. Specifically, the connecting bars (called tie-plates) between the individual cells were visually examined. Some of these tie-plates were observed to have a dimensional discrepancy to the design drawing when viewing the September 5, 2011 FOSAR video inspection. However, the comparison results from racks throughout the spent fuel pool show that these tie-plate images are the same in appearance both before and after the earthquake. These results provide high confidence that no significant physical or functional damage or distortion occurred to the spent fuel storage rack cells during the earthquake that would preclude the spent fuel storage racks from performing their intended design function.

Engineering has reviewed the manufacturer's design documentation for the North Anna Power Station Fuel Storage Racks, as well as pictures and videos which indicate that not all cell tie-plates are at perfect 90 degree angles as shown on the design drawings. Comparison of post and pre-seismic event videos shows these observed discrepancies existed prior to the earthquake. This as-found condition has been evaluated and determined to not compromise the ability of the storage racks to perform their structural design basis function.

The dummy fuel assembly was inspected and found to be in satisfactory condition, with no signs of any seismic damage. The dummy fuel assembly was then placed into 32 different spent fuel storage rack locations, which translates to two empty cells in each of the 16 spent fuel storage racks. The insertion and removal of the dummy assembly into these empty cells demonstrates that the spent fuel storage racks are functional and can support the insertion and removal of fuel assemblies using normal handling techniques.

Detailed video inspections were performed on the fuel during the pre-offload fuel pool shuffle (10 assemblies) and on five new fuel assemblies and Burnable Poison Rod Assemblies (BPRA) that were in the spent fuel storage racks during the earthquake. The irradiated fuel inspections were done using Dominion's normal criteria for fuel inspections, and the new fuel assembly inspections were done in accordance with the recommendations provided by AREVA. The results of these video inspections showed no anomalies or significant physical or functional damage due to the earthquake. These detailed video inspections provide assurance that there is no distortion of the fuel or BPRAs within the spent fuel storage racks. The fuel assembly video inspections also show that there was no evidence of any damage due to interaction between the fuel assembly and the storage cell. The pre-offload fuel pool shuffle consisted of more than

[1] Previous video exams from FOSARs, physical inventories or other pool activities were used to document the condition prior to the earthquake.

70 fuel assembly moves and was completed satisfactorily. Therefore, the detailed video inspections of these 15 fuel assemblies and the movement of other fuel assemblies as part of the pre-offload shuffle supports the conclusion that there is no degradation, rack deformation, or repositioning that could challenge the ability of the spent fuel racks to perform their intended function.

Conclusions

The inspections and testing that have been performed on fuel residing in the spent fuel storage racks, as well as on the racks themselves, show no discernable signs of earthquake-induced degradation or deformation that could challenge the ability of the spent fuel racks to perform their design functions. The current condition of the spent fuel storage racks has been evaluated and the ability of the racks to perform their structural design basis functions has been confirmed. Therefore, the spent fuel storage racks can be used for fuel movements, as allowed by current analyses and procedures. There are no spent fuel storage rack restrictions imposed as a result of this assessment.

References

1. "A Summary of Information in Support of Increasing the Spent Fuel Storage Capacity at North Anna Power Station Units 1 and 2," Attachment 3 to Letter from R. H. Leasburg (VEPCO) to H. R. Denton (USNRC), "Amendment to Operating Licenses NPF-4 and NPF-7, North Anna Power Station Unit Nos. 1 and 2, Proposed Technical Specification Changes," Serial Number 450, August 20, 1982.

2. NAPS UFSAR Section 9.1.2, "Spent-Fuel Storage."

3. Letter to USNRC (Document Control Desk) from VEPCO (L. N. Hartz), "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes, Increased Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit," dated September 27, 2000 (ADAMS No. ML003758403).

4. Letter to VEPCO (D. A. Christian) from USNRC (S. Monarque), "North Anna Power Station, Units 1 and 2 – Issuance of Amendments RE: Technical Specifications Changes to Increase Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit (TAC Nos. MB0197 and MB0198)," dated June 15, 2001 (ADAMS No. ML011700557).

Enclosure 6

Post-Earthquake Evaluation of the Independent Spent Fuel Storage Installation

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station**

Post-Earthquake Evaluation of the Independent Spent Fuel Storage Installation

Following the earthquake on August 23, 2011, the North Anna Independent Spent Fuel Storage Installation (ISFSI) was evaluated to determine if the earthquake had any detrimental effect on the spent fuel dry storage casks or the facility itself. The ISFSI is made up of two cask storage pads. Pad 1 is a long rectangular concrete pad upon which the dry storage casks stand vertically and positioned in pairs. Pad 1 contains Transnuclear TN-32 type storage casks. Pad 2 is a Nuclear Horizontal Modular Storage System (NUHOMS) where dry storage casks are placed horizontally in individual storage bunkers.

Design Criteria

The North Anna ISFSI Safety Analysis Report (SAR) for Pad 1, Section 3.2.3, defines the Design Basis Earthquake (DBE) peak ground acceleration values of 0.18g horizontal and 0.12g vertical for Seismic Class I structures founded on saprolite more than 15 feet thick. Both Pad 1 and Pad 2 are designed to these values. On Pad 1, the Transnuclear TN-32 casks were evaluated using the same design acceleration values as the pads. The TN-32 casks were evaluated for sliding at these values, and it was determined that sliding or tipping of the casks should not occur during a DBE.

On Pad 2, the NUHOMS Final Safety Analysis Report (FSAR), Section 2.2.3 states that the seismic design criteria for the NUHOMS HD System is based on NRC Regulatory Guide 1.60. The response spectra are anchored to a maximum ground acceleration of 0.30g for the horizontal components and 0.20g for the vertical component. The results of the frequency analysis of the Horizontal Storage Module (HSM)-H structure (which includes a simplified model of the dry storage cask) yield a lowest frequency of 23.2 Hz in the transverse direction and 28.4 Hz in the longitudinal direction. The lowest vertical frequency exceeds 33 Hz. Thus, based on the R.G. 1.60 response spectra amplifications, the corresponding seismic accelerations used for the design of the HSM-H are 0.37g and 0.33g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. The corresponding accelerations applicable to the dry storage casks are 0.41g and 0.36g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction.

Inspection Observations

Following the August 23, 2011 seismic event, two inspections of the ISFSI were conducted by the North Anna Fuel Handling team, Nuclear Analysis and Fuel, and Transnuclear personnel. The following conditions were observed at the ISFSI:

On Pad 1, an inspection was conducted on August 24, 2011 where it was observed that there were indications that twenty-five (25) of twenty-seven (27) TN-32 casks had moved slightly from their original placement locations. The pad is crowned in the middle with a 0.6 degree downward slope towards the ends; however, cask movement appeared to be independent of the slope of the pad. Directions of the cask movements are indicated in Table 1.

On a follow up inspection on September 1, 2011, location measurements of specific casks TN-32.42 and TN-32.32 were inconsistent with measurements taken on August 24, 2011. Measurements of TN-32.42 indicated a difference of 1.5 inches in measurement placing the cask 3 inches from its original placement. Measurements of TN-32.32 indicated a difference of 1 inch in measurement placing the cask 2.5 inches from its original placement. Match marks were added to the north and south of each cask and on the cask itself to provide a future reference of the placement locations of the TN-32 casks. Due to the limited data available, it is not possible to conclusively determine the cause of the discrepancy.

Inspections also revealed that the center-to-center spacing between twelve casks was less than the 16 feet nominal value provided in the ISFSI Technical Specifications. The two casks with the least separation are casks TN-32.16 and TN-32.23. The spacing of these casks measured 15 feet 3.5 inches during the inspections. The September 1, 2011 follow-up measurements identified an additional cask to be located less than 16 feet from another. Measurement of the distance between the outside trunnions of TN-32.32 and TN-32.42 was determined to be 15 feet 11.5 inches. The casks with spacing less than 16 feet between them following the seismic event are provided in Table 2.

The required spacing between TN-32 casks is specified in Section 4.2.3 of the North Anna ISFSI Technical Specifications (TS) and is a nominal 16 feet. TN-32 casks with a total decay heat above 27.1 kW require a minimum of 16 feet spacing. Of the thirteen casks that are closer than the 16 feet nominal value, two casks TN-32.38 and TN-32.48 were initially loaded with decay heat values exceeding 27.1 kW. Casks TN-32.38 and TN-32.48 were initially loaded with decay heat values of 28.7kW and 30.1kW, respectively. However, TN-32.38 was determined to have a decay heat of 24.3 kW as of January 1, 2008, and TN-32.48 was determined to have a decay heat of 25.7 kW as of January 1, 2009. Consequently, these two casks are currently below the 27.1 kW TS limit. The other TN-32 casks whose spacing is less than 16 feet were initially loaded with decay heats below 27.1kW.

On August 29, 2011 an inspection was conducted of the ISFSI with Transnuclear personnel present. During the inspection, visual inspection of the scrapes on the pad where the TN-32 casks had moved revealed paint that had come off of the bottom of the casks. The paint on the bottom of the casks is not credited for operability but is used to prevent corrosion of the carbon steel components. This observation was documented in a Condition Report.

Table 1: Pad 1 Cask Movement		
Cask No.	**8/24/2011 Measurement**	**9/1/2011 Measurement**
TN-32.49	E 2” and S 2.5”	E 2” and S 2.5”
TN-32.45	E 1.5”	E 1.5”
TN-32.43	E 2.5” and S 2.5”	E 2.5” and S 2.5”
TN-32.38	NW 1”	NW 1”
TN-32.37	SE 3”	SE 3”
TN-32.36	NW 4”	NW 4”
TN-32.29	S 2”	S 2”
TN-32.20	E 1”	E 1”
TN-32.23	SE 3.5”	SE 3.5”
TN-32.14	NW .75”	NW .75”
TN-32.16	NW 3.5”	NW 3.5”
TN-32.13	N 1”	N 1”
TN-32.12	NE .5”	NE .5”
TN-32.06	E 1.5”	E 1.5”
TN-32.10	SE 2.5”	SE 2.5”
TN-32.21	SE 4.5”	SE 4.5”
TN-32.19	E 2.25”	E 2.25”
TN-32.24	E 3”	E 3”
TN-32.26	E 1.25”	E 1.25”
TN-32.32	SE 1.5”	SE 2.5”
TN-32.30	No Movement	No Movement
TN-32.41	No Movement	No Movement
TN-32.42	SE 1.5”	SE 3”
TN-32.47	S 1”	S 1”
TN-32.48	NW 2”	NW 2”
TN-32.52	NW 2”	NW 2”
TN-32.53	SE 2.5”	SE 2.5”

Table 2: Pad 1 Casks Center-to-Center Spacing Measurements (from Outside Trunnions)		
Cask Nos.	**8/24/11 Measurement**	**9/1/11 Measurement**
TN-32.13 and TN-32.10	15'-6"	15'-6"
TN-32.16 and TN-32.23	15'-3.5"	15'-3.5"
TN-32.26 and TN-32.30	15'-10"	15'-10"
TN-32.42 and TN-32.48	15'-11"	15'-10.5"
TN-32.06 and TN-32.12	15'-11.5"	15'-11.5"
TN-32.45 and TN-32.38	15'-11"	15'-11"
TN-32.32 and TN-32.42	16'	15'-11.5"

Above ground pressure monitoring systems including the remote monitoring panel were visually inspected and no significant physical or functional damage was found. No pressure monitoring system alarms were received during or subsequent to the event. No loss of electrical power at the ISFSI occurred during the event. Visual inspection of the pad itself did not reveal any cracking or damage from the seismic event. Visual inspections of the TN-32 casks also did not reveal any significant physical or functional damage to the casks. Radiological surveys of the casks on Pad 1 indicated no changes to the cask surface dose. Checks were done on six randomly selected pressure switches on August 30, 2011. It was verified during these inspections that setpoints had not drifted outside of what would normally be expected and helium pressure was found to be consistent with expectations.

An inspection of Pad 2 was conducted on August 24, 2011. Observations are documented below and also include supplemental information obtained during a subsequent inspection performed with Transnuclear personnel on August 29, 2011. From these inspections, it was determined that the only damage observed on the Horizontal Storage Modules (HSMs) would not impact the structural integrity or radiation shielding capability of the HSMs. With the exception of HSM 01 and HSM 03, damage to the HSM array was limited to unloaded HSMs. The spaces between some HSMs indicated that some minor movement had occurred with a maximum gap between HSM 24 and HSM 26 of 1.5 inches at the top. The inspections identified the following items, which were documented in a Condition Report:

Table 3: Pad 2 Observations	
Components	**Description**
HSM 24-HSM 26	Side by side gap was .75" at the bottom and 1.5" at the top
HSM 24-HSM 26	Inlet bird screen indicated buckling
HSM 24-HSM 26	Outlet vent cover appears to have moved pulling the bird screen away and damaging concrete.
HSM 06-HSM 08	Side by side gap was .75" from bottom to top
HSM 06-HSM 04	Roof gap .5"
HSM 13-HSM 15	Roof gap 1.125"
HSM 15-HSM 17	Roof vent had broken concrete with exposed rebar and bird screen has been pulled away 3/16"
HSM 18-HSM 20	Outlet vent cover cracked
HSM 3	Cracked/chipped concrete at inlet screen that extended behind bird screen
HSM 1	Minor crack on lower front of the base
HSM 25	Concrete loose on left rear top corner
HSM 23- HSM 25	Outlet vent cover is cracked
Handrails	Moved 0.5" N

In addition to the observations in Table 3, on a follow-up inspection the HSM fasteners on one roof vent cover and on the south west end wall were discovered to be loose. A work order was requested to inspect and tighten all fasteners as required. The loaded HSMs have been determined to be capable of performing their intended design functions, and radiological conditions around the HSMs are normal. Damage to the HSMs will need to be repaired prior to loading. Cask movement has been restricted to prevent loading the HSMs prior to repair. Visual inspection of cracks in the pad indicated they were preexisting and unrelated to the seismic event.

<u>Conclusions</u>

Following the seismic event at North Anna, there were indications of cask movement. On Pad 1, there were indications the TN-32 casks had moved and center-to-center spacing between thirteen casks is below the 16 feet nominal value. There were no indications of significant physical or functional damage to the casks. On Pad 2, two loaded HSMs indicated minor cracks on the bases. The majority of the damage was to unloaded HSMs; however, no significant physical or functional damage was identified, and it will not affect the HSMs ability to perform their intended functions. Radiological conditions remained unchanged for both pads at the ISFSI.

Open issues associated with the ISFSI have no impact on the return to service of Unit 1 or Unit 2 and are being tracked in the Corrective Action System. Finally, Dominion has concluded that the TN-32 casks and the NUHOMS HD systems continue to perform their shielding, criticality, thermal, and confinement design functions.

Enclosure 7

Post-Earthquake Impact Assessment on Engineering Programs

Virginia Electric and Power Company
(Dominion)
North Anna Power Station

Impact Assessment of August 23, 2011 Earthquake on Engineering Programs

The purpose of this assessment is to evaluate the impact of the August 23, 2011 seismic event at North Anna Power Station on Engineering Programs through the review of Dominion procedures, regulatory documents, and industry related source documents for each of the identified functional areas to determine if plant equipment or supporting documentation requires additional analysis or inspection in response to exceeding either the Operating Basis Earthquake (OBE) or the Design Basis Earthquake (DBE).

The Engineering Programs that were reviewed in detail as part of this assessment include the following:

1. Aging management,
2. Air Operated Valves,
3. Generic Safety Issue (GSI)-191, Containment Sump Performance,
4. Environmental Qualification,
5. Fire Protection/Appendix R,
6. Heat Exchangers (Generic Letter 89-13),
7. Inservice Inspection Program,
8. Inservice Inspection Program - Containment,
9. Inservice Inspection Program - Repair and Replacement,
10. Inservice Inspection Program - System Pressure Tests,
11. Motor Operated Valves,
12. Maintenance Rule Program Compliance,
13. Steam Generators,
14. Reactor Vessel and Internals/Reactor Coolant System, and
15. Buried Pipe Monitoring/Ground Water Monitoring Program.

A review of the Dominion, regulatory, and industry documents for each of the identified functional areas has been completed based upon the magnitude of the earthquake exceeding the DBE. As a result of these reviews, it was concluded that additional program actions are necessary for certain Engineering Programs beyond the guidance given in EPRI Technical Report NP-6695. The affected Engineering Programs include the following functional areas:

1. Steam Generators,
2. GSI-191, Containment Sump Performance,

3. Inservice Inspection (ISI),
4. Buried Pipe Monitoring/Ground Water Monitoring Program,
5. Reactor Vessel and Internals/Reactor Coolant System,
6. ISI - Repair & Replacement, and
7. Aging Management.

The following actions were identified and will be completed prior to unit restart based on exceeding the spectral and peak ground accelerations of the DBE.

Steam Generators: EPRI Steam Generator Management Program Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7, Section 3.10 states that forced outage examinations shall be performed during plant shutdown subsequent to seismic occurrence greater than the OBE.

Perform a 20% sample inspection of Unit 1 "A" and Unit 2 steam generators. The Unit 1 "A" steam generator inspection is in progress, and, to date, no adverse indications have been identified as a result of the seismic event. The Unit 2 "A" and "C" steam generators will be inspected during the current refueling outage as previously planned.

GSI-191, Containment Sump Performance
1. Perform containment inspections to identify and remove debris that may have resulted from the earthquake, as required.
2. Perform a visual examination of the sump strainer gaps in accordance with the applicable periodic test.

Inservice Inspection
Perform sample weld inspections of reactor coolant loop drain lines, service water tie-in vault, and penetration area pipe lines with anchors.

Buried Pipe Monitoring/Ground Water Monitoring Program
Perform buried pipe inspections of:
- the two areas of buried fire protection pipe that are currently excavated,
- the Unit 2 circulating water discharge tunnel and associated liquid waste line, and
- the buried pipe between the Unit 1 auxiliary feedwater tunnel and the Unit 1 Quench Spray Pump House.

Enclosure 8

Near-Term Actions to be Completed Prior to Unit Restart

Virginia Electric and Power Company
(Dominion)
North Anna Power Station

NEAR-TERM ACTIONS TO BE COMPLETED PRIOR TO UNIT RESTART		
	Restart Activity	**Comments**
A. Seismic Monitoring and Design Basis		
1	Provide temporary backup power to the Main Control Room Seismic Monitoring Panel.	Complete
2	Install temporary free field seismic monitoring instrumentation.	Prior to Unit 1/2 Restart
3	Revise Abnormal Procedure 0-AP-36 to improve procedural guidance for determining whether an onsite earthquake exceeds OBE and/or DBE peak acceleration criteria.	Prior to Unit 1/2 Restart
B. Nuclear Fuel		
1. Unit 1 Core		
a	Perform hot rod drop testing.	Prior to Unit 1 entering Mode 2
2. Unit 2 Core		
a	Perform RCCA drag testing.	Prior to Unit 2 onload
b	Perform hot rod drop testing.	Prior to Unit 2 entering Mode 2
c	Perform routine binocular visual inspection during core offload.	Prior to Unit 1 Restart
d	Perform video inspections on 13 benchmark assemblies and additional vendor-recommended assemblies.	Prior to Unit 1 Restart
e	Perform video inspection of RCCA hubs.	Prior to Unit 1 Restart
f	Perform video inspections on assemblies with anomalies observed during binocular inspections.	Prior to Unit 1 Restart
C. Root Cause Evaluations		
1	Reactor Trip	Prior to Unit 1/2Restart

NEAR-TERM ACTIONS TO BE COMPLETED PRIOR TO UNIT RESTART		
Restart Activity		**Comments**
2	Unit 2H Emergency Diesel Generator Coolant Leak	Prior to Unit 1/2 Restart
D. Inspections		
1	Steam Generators - Perform a 20% sample inspection of Unit 1 and Unit 2 steam generators.	Prior to Unit 1/2 Restart
2	Containment - Perform containment inspections to identify and remove debris that may have resulted from the earthquake, as required.	Prior to Unit 1/2 Restart
3	Containment Sump Strainers Perform a visual examination of the sump strainer gaps in accordance with the applicable periodic test.	Prior to Unit 1/2 Restart
4	Inservice Inspection Perform sample weld inspections.	Prior to Unit 1/2 Restart
5	Buried Pipe Monitoring/Ground Water Monitoring Program Perform buried pipe inspections of: • the two areas of buried fire protection pipe that are currently excavated, • the Unit 2 circulating water discharge tunnel and associated liquid waste line, and • the buried pipe between the Unit 1 auxiliary feedwater tunnel and the Unit 1 Quench Spray Pump House.	Prior to Unit 1/2 Restart
E. Testing		
1	Complete Unit 1/2 Surveillance Periodic Tests as determined by the Seismic Event Response Team.	Prior to and during Unit 1/2 Startup per Technical Specifications (Unit specific tests will be completed prior to and during that Unit's startup)

Enclosure 9

Long-Term Actions to be Completed After Unit Restart

**Virginia Electric and Power Company
(Dominion)
North Anna Power Station**

LONG-TERM ACTIONS TO BE COMPLETED AFTER UNIT RESTART	
Activity	
A. Seismic Monitoring and Evaluations	
1	Provide permanent backup power to the Main Control Room Seismic Monitoring Panel.
2	Install permanent free field seismic monitoring instrumentation.
3	Reevaluate plant equipment identified in the IPEEE review with HCLPF capacity <0.3g.
4	Perform seismic evaluations in the context of EPRI NP-6695, NRC GI-199 and as an outcome of NRC Task Force recommendations identified in SECY-11-0124.
B. Reactor Vessel Internals	
1	Develop a plan with the NSSS vendor consisting of additional evaluations or inspections, as warranted, to assure long term reliability of the reactor internals.

Baggett, Steven

From: Gilles, Nanette
Sent: Thursday, August 25, 2011 8:34 AM
To: Apostolakis, George; Baggett, Steven; Sosa, Belkys
Subject: Fw: one pager for chairman on North Anna
Attachments: 1 Pager for Chairman Jaczko on North Anna Earthquake Issue.docx

FYI

Sent from my NRC Blackberry

From: Merzke, Daniel EDO
To: Monninger, John
Cc: Hipschman, Thomas; Marshall, Michael; Castleman, Patrick; Gilles, Nanette; Orders, William; Franovich, Mike; Ash, Darren
Sent: Thu Aug 25 07:21:06 2011
Subject: FW: one pager for chairman on North Anna

John, here is a one-pager for the Chairman on the analysis of the seismic activity at North Anna. I hope this "hits the mark." Let me know if there's something else he was looking for.

Dan

From: Wilson, George NRR
Sent: Thursday, August 25, 2011 5:46 AM
To: Grobe, Jack; Boger, Bruce; Leeds, Eric; Ruland, William; McGinty, Tim; Lund, Louise; Pruett, Troy; Lubinski, John; Wiggins, Jim; Dapas, Marc; McCree, Victor; Croteau, Rick; Jones, William; Giitter, Joseph; Howe, Allen; Evans, Michele; Holian, Brian; Skeen, David; Galloway, Melanie; Cheok, Michael; Nelson, Robert; Bahadur, Sher; Andersen, James; Dean, Bill; Virgilio, Martin; Borchardt, Bill; Weber, Michael; Johnson, Michael; Holahan, Gary; Merzke, Daniel; Sanfilippo, Nathan; Hayden, Elizabeth; Chokshi, Nilesh; Wert, Leonard; Hiland, Patrick; Skeen, David
Cc: Li, Yong; Karas, Rebecca; Khanna, Meena; Munson, Clifford; Kammerer, Annie; Manoly, Kamal; Wertz, Trent; Martin, Robert; Thomas, George; Taylor, Robert
Subject: one pager for chairman on North Anna

The attached is the requested one page write up on North Anna from the Chairman's office

George Wilson
USNRC
EICB Branch Chief, Division of Engineering
Mail Stop O12H2
301-415-1711

L18

EDO | NRR

Summary of Earthquake Information for the North Anna NPP as of August 24, 2011

The North Anna Nuclear Power Plant (NANPP) has two Safe Shutdown Earthquake (SSE) ground motions, one for structures, systems, and components (SSCs) located on top of rock, which is anchored at 0.12 g, and the other is for SSCs located on top of soil, which is anchored at 0.18 g. The NANPP has two corresponding Operating Basis Earthquake (OBE) ground motion spectra, anchored at 0.09 g for soil and 0.06 g for rock. The figure below shows a comparison between the Safe Shutdown Earthquake (SSE) and OBE for Units 1 and 2, the Unit 3 Combined License (COL) application Ground Motion Response Spectrum (GMRS), the current best estimate of the August 24, 2011 earthquake ground motions from the USGS (ShakeCast version 6), and predicted median and standard deviation earthquake motions using the EPRI ground motion prediction equations. The IPEEE review ground motion (not shown) was anchored at 0.16 g with a similar spectrum as the SSE.

The recent earthquake occurred at a close distance to the plant with a magnitude of 5.8 at a relatively shallow depth. USGS estimates of the maximum ground motion at the plant evolved as new data become available. The current best estimate of the Peak Ground Acceleration (PGA) for the NANPP site is 0.2g, which contains uncertainty and may be updated later. This estimate indicates that the ground motion likely exceeded the SSE response spectra for NANPP Units 1 and 2 (0.12g) over a considerable frequency range, as shown by the green and red points in the figure. The estimated ground motion from the earthquake was not a surprise based on the combined operating license application (COLA) ground motion response spectrum for NANPP Unit 3. This preliminary estimate appears to validate the NRC's current seismic hazard assessment approaches and models for new reactors, as well as the basis for GI-199 reviews.

The USGS ground motion estimate values for the plant site are developed based on two types of input. The principal input are theoretical predicted ground motions that come from analyses in which recorded motions at seismograph stations are extended to the NPP sites using ground motion prediction equations (also called attenuation relationships). This theoretical prediction is then modified based on intensity information that comes from the USGS "Did You Feel It?" (DYFI) system. The DYFI system is a method for using large numbers of inputs from affected persons to develop intensity maps that are used as a "ground truth." Currently, the USGS has received nearly 123,000 submitted reports.

NRC staff performed an independent analysis using the best estimate of the earthquake location and magnitude together with the EPRI ground motion prediction equations. The median and ±1 standard deviation curves are shown. It can be seen that the 84^{th} percentile ground motions calculated by the staff are close to the USGS predictions. This makes sense because the USGS theoretical values were increased due to the intensity information that came out of the DYFI system.

Currently, the licensee is retrieving its seismic instrumentation recordings. However, we do not yet know the type and quality of the recording data that will be available to the NRC. Information from the NANPP will be used to evaluate the USGS estimates of ground motion and will be compared against the FSAR design basis. The data will be used to inform the staff whether additional analysis is needed.

The licensee is expected to perform plant walk downs in accordance with RG 1.167, "Restart of a Nuclear Power Plant Shutdown by a Seismic Event," which endorses EPRI's "Guidelines for Nuclear Plant Response to an Earthquake" with conditions. If the SSE is exceeded at certain frequencies, the staff will assess the licensee's evaluation of SSCs that are most sensitive to ground motion in that frequency band.