Hoxie, Chris

Ok to release

From: Hoxie, Chris
Sent: Saturday, March 26, 2011 9:42 PM
To: Uhle, Jennifer
Cc: Lee, Richard; Gibson, Kathy
Subject: Brian's Q

In regards to Brian's question about how salt water may influence the dynamics of a fuel coolant interaction:

Here are two references:
http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/42/006/42006251.pdf

On page 396, states that pure water vs. salt water made no difference in an experiment designed to measure peak pressures for fuel coolant interactions in a lab setting.

Reference 2:
Although it might not be one-for-one, here is a reference to research that indicates the salt might actually dampen the steam explosion (or at least it does maybe when lava hits sea water....)

Caveats: This reference 2 is not nuclear oriented. Not specific to the Japan case. This is really complex and should be answered by an expert. Depends so much on the actual conditions in the Japan plants...

At least I did not find anything that says salt makes things worse!

Impure coolants and interaction dynamics of phreatomagmatic eruptions

James D. L. WhiteE-mail The Corresponding Author, *

Geology Department, University of Otago P.O. Box 56, Dunedin 9015, New Zealand
Received 12 January 1996;
revised 18 June 1996;
accepted 18 June 1996. ;
Available online 26 February 1999.

Abstract

Phreatomagmatic eruptions resulting from interaction of magma with groundwater are common in many terrestrial settings, and their explosivity is widely accepted to result from fuel-coolant interaction (FCI) processes. Relatively little attention has been given to the precise nature of the volcanic settings in which phreatomagmatic FCI's take place, but several lines of evidence indicate that they almost inevitably involve mixing of magma with impure, sediment-laden water. Consideration of the effects of these impure coolants on the fuel-coolant interaction process suggests that: (1) impure coolants enhance the ability of magma to mix with large volumes of coolant; and **(2) maximum unit-volume explosivity of FCI's is damped relative to interactions with pure water.** It is probably unrealistic to back-calculate water-magma mass ratios for most, if not all, phreatomagmatic eruptions because: (1) effects of impure coolants on fragmentation efficiency and eruption explosivity are not yet known; and (2) aspects of the vent environments in which phreatomagmatism occurs may influence fragmentation processes, explosive efficiency, and resultant particle populations as or more strongly than water-magma mass ratios. To estimate mass ratios for individual bursts, or for eruptions as a whole, one must distinguish particle populations resulting from many different processes in phreatomagmatic vents, including primary fragmentation, induced fragmentation, vent-wall collapse and pyroclast recycling. Incorporation of accidental blocks beyond the zone of phreatomagmatic interaction and ejection of unvaporized water further complicate efforts at reconstruction.

4/285

Schaperow, Jason

Subject: Support for Fukushima accident
Location: Charlie's office

Start: Mon 3/28/2011 10:00 AM
End: Mon 3/28/2011 11:00 AM
Show Time As: Tentative

Recurrence: Daily
Recurrence Pattern: every day from 10:00 AM to 11:00 AM

Meeting Status: Not yet responded

Organizer: Schaperow, Jason
Required Attendees: Esmaili, Hossein; Salay, Michael; Marksberry, Don; Helton, Donald; Tinkler, Charles

Request you come to Charlie's office at 10:00 a.m. to meet.

W/286

Schaperow, Jason

From: Schaperow, Jason
Sent: Monday, March 28, 2011 2:05 PM
To: Greenwood, Carol
Subject: RE: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

Thanks.

From: Greenwood, Carol
Sent: Monday, March 28, 2011 1:54 PM
To: Schaperow, Jason
Subject: RE: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

Yes, Thank you!

Carol

From: Schaperow, Jason
Sent: Monday, March 28, 2011 1:53 PM
To: Greenwood, Carol
Cc: Santiago, Patricia
Subject: RE: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

Hi Carol,

I got a call from the Ops Center yesterday morning at 0600 for support. So, I worked from 0600 to 0800 yesterday. I used the link below to add this to your timesheet. Did I do it correctly?

Thanks,
Jason

From: Greenwood, Carol
Sent: Friday, March 18, 2011 10:31 AM
To: Armstrong, Kenneth; Bajorek, Stephen; Boyd, Christopher; Elkins, Scott; Hoxie, Chris; Lee, Richard; Rubin, Stuart; Santiago, Patricia; Sherbini, Sami; Tinkler, Charles; Voglewede, John; Zigh, Ghani; Tomon, John
Subject: FW: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

Please update the Excel spreadsheet by clicking here with names and dates of any staff that have or will be performing emergency-related premium work in response to the events in Japan.
This applies to the IRC, OIP, OPA or wherever they are doing emergency work.

Please confirm to me when your branch is updated.

The spreadsheet is at g:\DSA\Directors Office\JapanResponseWork.xlsx if the above link doesn't work.

Regards
Carol Greenwood
Lead Administrative Assistant

4287

RES/DSA
U.S. Nuclear Regulatory Commission
Phone: 301-251-3319

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

From: Gibson, Kathy
Sent: Friday, March 18, 2011 8:07 AM
To: Greenwood, Carol
Subject: Fw: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

Would you check with the BCs and compile this list for Andrea for DsA? Thx

---

From: Valentin, Andrea
To: Gibson, Kathy; Scott, Michael; Coyne, Kevin
Sent: Fri Mar 18 08:00:34 2011
Subject: Reminder: FW: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

This is a reminder to provide me with a list of names of staff that are performing emergency-related premium work (and the dates that the people worked) in response to the events in Japan. This applies to the IRC, OIP, OPA or wherever they are doing emergency work.

Thanks,
Andrea

From: Khan, Charline
Sent: Thursday, March 17, 2011 7:29 AM
To: RidsAcrsAcnw_MailCTR Resource; RidsAslbpManagement Resource; RidsOgcMailCenter Resource; RidsOcaaMailCenter Resource; RidsOcfoMailCenter Resource; RidsOigMailCenter Resource; RidsOipMailCenter Resource; RidsOcaMailCenter Resource; RidsOpaMail Resource; RidsSecyMailCenter Resource; RidsSecyCorrespondenceMCTR Resource; RidsEdoMailCenter Resource; RidsAdmMailCenter Resource; RidsCsoMailCenter Resource; RidsOeMailCenter Resource; RidsFsmeOd Resource; RidsOiMailCenter Resource; RidsOIS Resource; RidsHrMailCenter Resource; RidsNroOd Resource; RidsNroMailCenter Resource; RidsNmssOd Resource; RidsNrrOd Resource; RidsNrrMailCenter Resource; RidsResOd Resource; RidsResPmdaMail Resource; RidsSbcrMailCenter Resource; RidsNsirOd Resource; RidsNsirMailCenter Resource; RidsRgn1MailCenter Resource; RidsRgn2MailCenter Resource; RidsRgn3MailCenter Resource; RidsRgn4MailCenter Resource
Cc: Davidson, Lawrence; Buchholz, Jeri; Johns, Nancy
Subject: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

MEMORANDUM TO: Those on the Attached List

FROM: Miriam L. Cohen, Director**/RA by J. Buchholz for/**
Office of Human Resources

DATED: March 16, 2011

SUBJECT: WAIVER OF WORK SCHEDULE AND PAY CAP RULES FOR WORK IN RESPONSE TO THE EVENTS IN JAPAN

ADAMS Accession No. ML11075A003 refers

<u>NOTE: Electronic distribution only</u>

Charline Khan
Administrative Assistant (Rotation)
U.S. NUCLEAR REGULATORY COMMISSION
Office of Human Resources
P:301-492-2318
Charline.Khan@nrc.gov

Releasable

Bensi, Michelle

From: Bensi, Michelle
Sent: Monday, March 28, 2011 2:35 PM
To: Guzman, Richard
Subject: RE: Confirmation on Pilgrim info (site fact sheet for governor)

Richard,
This is great. Thank you very much.
Shelby

From: Guzman, Richard
Sent: Monday, March 28, 2011 1:18 PM
To: Bensi, Michelle
Subject: RE: Confirmation on Pilgrim info (site fact sheet for governor)

Shelby,

Both items are addressed in the FSAR as follows:

- The seismic design standard for Pilgrim safety-related equipment was determined by applying the effects of the largest earthquake in the region, an event measuring 6.0 on the Richter scale, at Cape Anne, 60 miles north. This event is then applied at the closest epicentral location consistent with geologic structure of the site.

2.5.3.3.2 Safe Shutdown Earthquake

The Safe Shutdown Earthquake is generally considered to be a recurrence of the largest earthquake in the region at the closest epicentral distance which is consistent with the geologic structure. The Cape Ann series of earthquakes appear to be the most severe earthquakes which need be considered for plant design. The occurrence of an earthquake as large as the maximum Cape Ann sequence (intensity VIII, estimated magnitude 6), with its epicenter at the closest approach of faulting associated with the Boston and Narragansett Basins (17 mi west of the site) is the most critical situation for the site. Horizontal ground acceleration at estimated foundation depths (within the compact glacial deposits) due to the above earthquake would be about 0.15 g.

- A tsunami at Pilgrim such as occurred in Japan is not considered to be a probable event based on the known geological features in the area. The emergency diesel generators that provide power if the site loses off-site power ~~and~~ are built in reinforced concrete watertight buildings and the fuel tanks are built underground in reinforced concrete.

FSAR pg 8.5-4 – "Both generators are housed in <u>reinforced concrete Class I structures</u>. Each unit is completely enclosed to provide independence from the other unit."

FSAR pg 8.5-8 – "Each diesel generator is capable of starting and continuously operating at full rated capacity for a period of 7 days using fuel stored onsite in <u>underground storage tanks</u>."

I've also attached the applicable FSAR sections for additional information. Hope this helps!

Have a good one,
Rich

Rich Guzman

4288

Sr. Project Manager
NRR/DORL
US NRC
301-415-1030
Richard.Guzman@nrc.gov

From: Bensi, Michelle
Sent: Monday, March 28, 2011 12:15 PM
To: Guzman, Richard
Subject: RE: Confirmation on Pilgrim info (site fact sheet for governor)

Richard,
I just found on that the SLO wants to give this information to the MA governor by the end of the day. The appointment between the NRC and SLO for the governor is at 2pm today (at which time we'd like to have "fact-checked" everything), so there's a bit of a time crunch on things.
Thanks again,
Shelby

From: Bensi, Michelle
Sent: Monday, March 28, 2011 11:59 AM
To: Guzman, Richard
Subject: Confirmation on Pilgrim info (site fact sheet for governor)

Hi Richard,

Thanks for taking the time to talk with me a few minutes ago.

The Massachusetts SLO has pulled together a factsheet for the governor. We've been asked to review it for errors. Here are the two "facts" I need to confirm relative to Pilgrim (extracted directly out of document):

- The seismic design standard for Pilgrim safety-related equipment was determined by applying the effects of the largest earthquake in the region, an event measuring 6.0 on the Richter scale, at Cape Anne, 60 miles north. This event is then applied at the closest epicentral location consistent with geologic structure of the site.
- A tsunami at Pilgrim such as occurred in Japan is not considered to be a probable event based on the known geological features in the area. The emergency diesel generators that provide power if the site loses off-site power and built in reinforced concrete watertight buildings and the fuel tanks are built underground in reinforced concrete.

I know you are busy, but I'd really appreciate it if you could get back to me quickly on these two points.

Thanks again,
Shelby

Michelle Bensi, Ph.D.
Reliability and Risk Engineer
Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Risk Analysis
Operating Experience and Generic Issues Branch

Bensi, Michelle

From: Bensi, Michelle
Sent: Monday, March 28, 2011 3:02 PM
To: OST02 HOC; OST01 HOC
Subject: RE: RST Support Seismology Q&A position

Hello,
I was brought in primarily to assist with compilation of a seismic Q&A document and I continue to work on that this week. Thus, I wasn't planning to work any shifts in the Ops Center this week. Please let me know if this is a problem.
Thanks,
Michelle Bensi

From: OST02 HOC
Sent: Friday, March 25, 2011 4:34 PM
To: Weaver, Thomas; Munson, Clifford; Seber, Dogan; Devlin, Stephanie; Bensi, Michelle
Subject: RST Support Seismology Q&A position

Please designate which shifts this weekend and next week, starting 7:00am, tomorrow morning, March 26th, for Seismology Q&A questions . Send responses back to OST01. HOC@nrc.gov, OST02.HOC@nrc.gov.

EST Admin Support
NRC Operations Center
eMail: OST02.HOC@nrc.gov

W/289

Schaperow, Jason

From: Schaperow, Jason
Sent: Monday, March 28, 2011 4:10 PM
To: Chang, Richard
Subject: RE: SOARCA 2011 RIC Slides

Thank you.

From: Chang, Richard
Sent: Monday, March 28, 2011 3:21 PM
To: Dacus, Eugene
Cc: Sheron, Brian; Schaperow, Jason; Armstrong, Kenneth; Gibson, Kathy; Wagner, Katie
Subject: SOARCA 2011 RIC Slides

Eugene,

Here is the link for the SOARCA session at the 23rd Regulatory Information Conference.

https://ric.nrc-gateway.gov/docs/abstracts/SessionAbstract_58.htm

Please let me know if there is anything else that I can help you with.

Richard Chang
Program Manager
RES/DSA/SPB
301-251-7980

U/290

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, March 29, 2011 9:40 AM
To: Chang, Richard
Subject: RE: FYI- News Article on SOARCA

Thanks.

From: Chang, Richard
Sent: Tuesday, March 29, 2011 7:35 AM
To: Schaperow, Jason; Tinkler, Charles; Santiago, Patricia; Ghosh, Tina; Armstrong, Kenneth
Subject: FYI- News Article on SOARCA

http://news.yahoo.com/s/ap/20110329/ap_on_re_us/us_us_japan_nuclear_blackouts_2

Richard Chang
Program Manager
RES/DSA/SPB
301-251-7980

4/291

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, March 29, 2011 9:51 AM
To: Chang, Richard
Subject: RE: SOARCA Peer Review Committee

Sounds good to me.

From: Chang, Richard
Sent: Tuesday, March 29, 2011 8:31 AM
To: Tinkler, Charles; Schaperow, Jason
Cc: Ghosh, Tina
Subject: SOARCA Peer Review Committee

Guys,

I am planning on writing the Peer Review Committee an e-mail stating that the events in Japan have delayed the release of Appendix A to them by an as-of-yet undetermined amount of time (and that an estimate will not be available until the reactors in Japan stabilize).

Do you have any thoughts on that?

Thanks,

Richard Chang
Program Manager
RES/DSA/SPB
301-251-7980

1292

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, March 29, 2011 11:42 AM
To: Lee, Richard
Subject: RE: DOE trip to Milestone

Thanks. Very interesting.

From: Lee, Richard
Sent: Tuesday, March 29, 2011 11:20 AM
To: Esmaili, Hossein; Gauntt, Randy (home); Randy Gauntt (SNL); Salay, Michael
Cc: Tinkler, Charles; Schaperow, Jason; Katie Wagner
Subject: DOE trip to Milestone

Enclosed is a brief trip report from Per Peterson on DOE trip to Milestone yesterday.

Katie: Please log in on share point. It is provided to DSA staff for information.

4/293

Lee, Richard

From: Lee, Richard
Sent: Tuesday, March 29, 2011 9:39 PM
To: Powers, Dana A
Subject: RE: gauntt to japan

He will pay for this. Have you able to get one of his staff to answer the questions. Larry and some can help.

From: Powers, Dana A [dapower@sandia.gov]
Sent: Tuesday, March 29, 2011 7:30 PM
To: Lee, Richard
Subject: gauntt to japan

I take it you have heard that Randy is being sent to Japan by SNL! He is just trying to get out of preparing response to the peer reviewers. Dana

11/294

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Tuesday, March 29, 2011 9:24 AM
To: McNamara, Nancy
Subject: RE: Briefing Package for MA Visit & 11:00 call
Attachments: Fukushima Presentation (3-25) with GI199.pptx

Nancy,

I took the liberty of adding the GI-199 slides to the Fukushima presentation so that they have the same look and can be together if you are printing handouts. I also decided to swap the position of GI-199 slides 3 and 4. The revised presentation is attached for your use.

Ben

From: McNamara, Nancy
Sent: Tuesday, March 29, 2011 8:59 AM
To: Beasley, Benjamin; Schmidt, Wayne
Subject: Briefing Package for MA Visit & 11:00 call
Importance: High

4295

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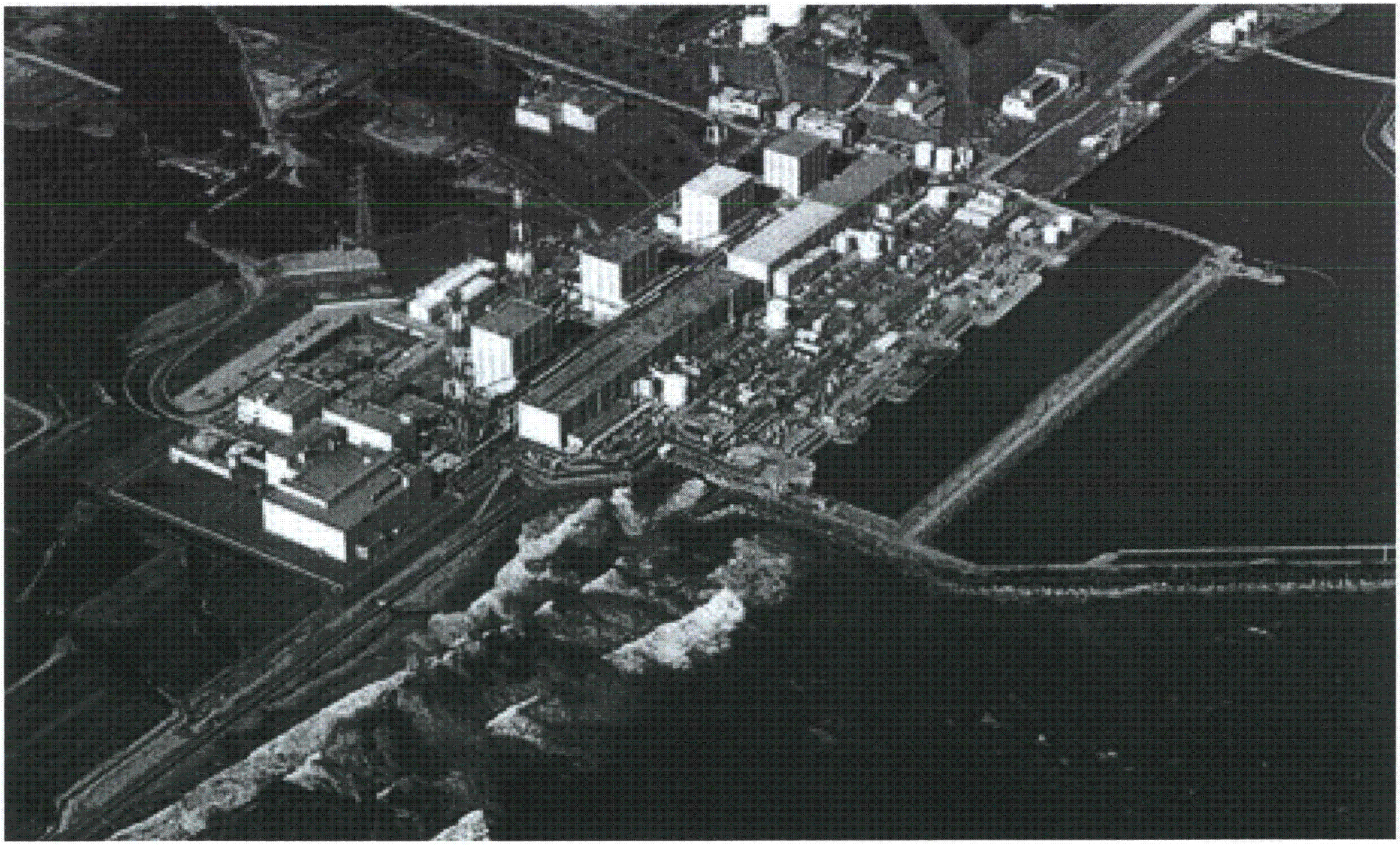
Events at Fukushima Units 1-4

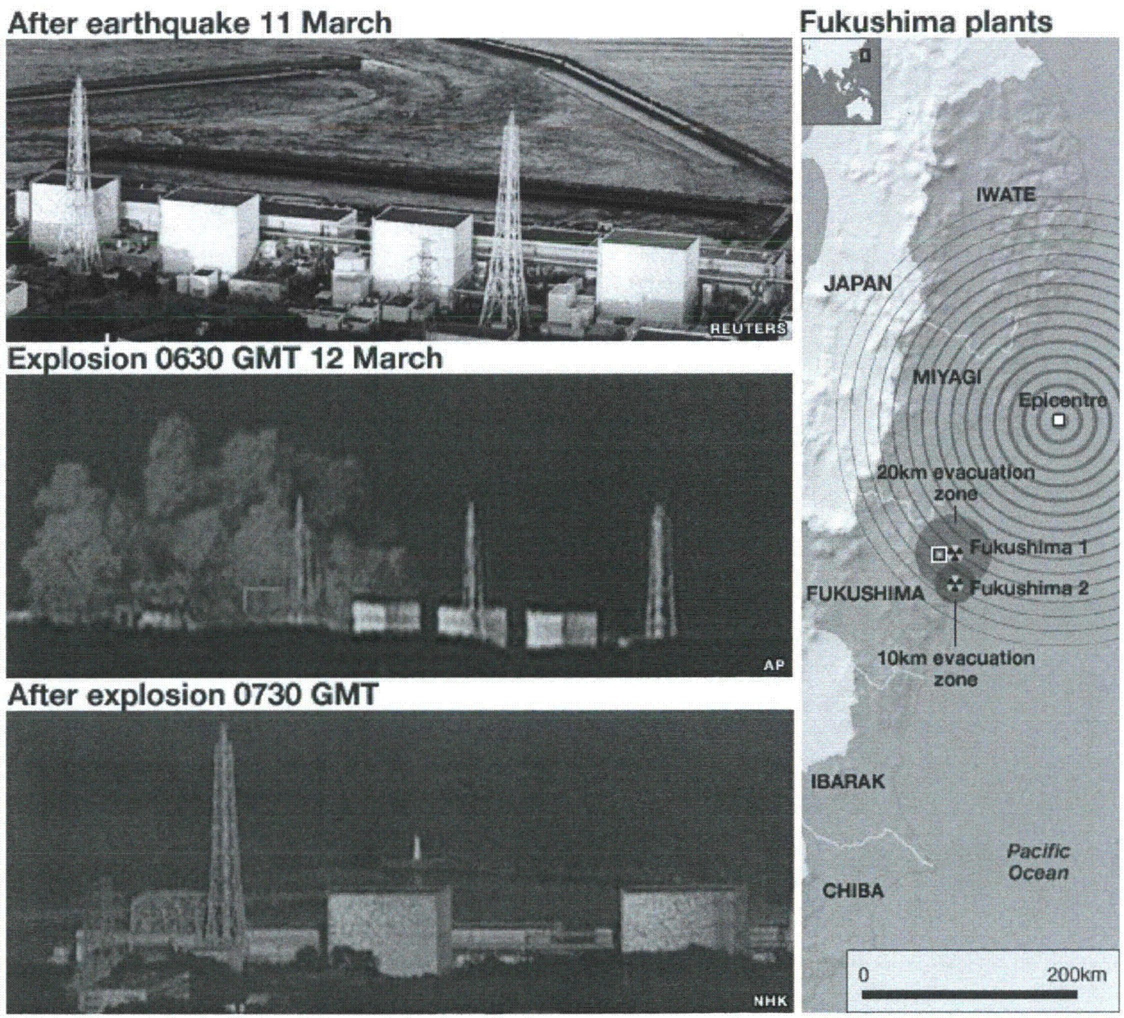
March 30, 2011

Bill Dean, Regional Administrator

Fukushima Units 1 - 4



3/11 Earthquake & 3/12 Unit 1 Hydrogen Explosion



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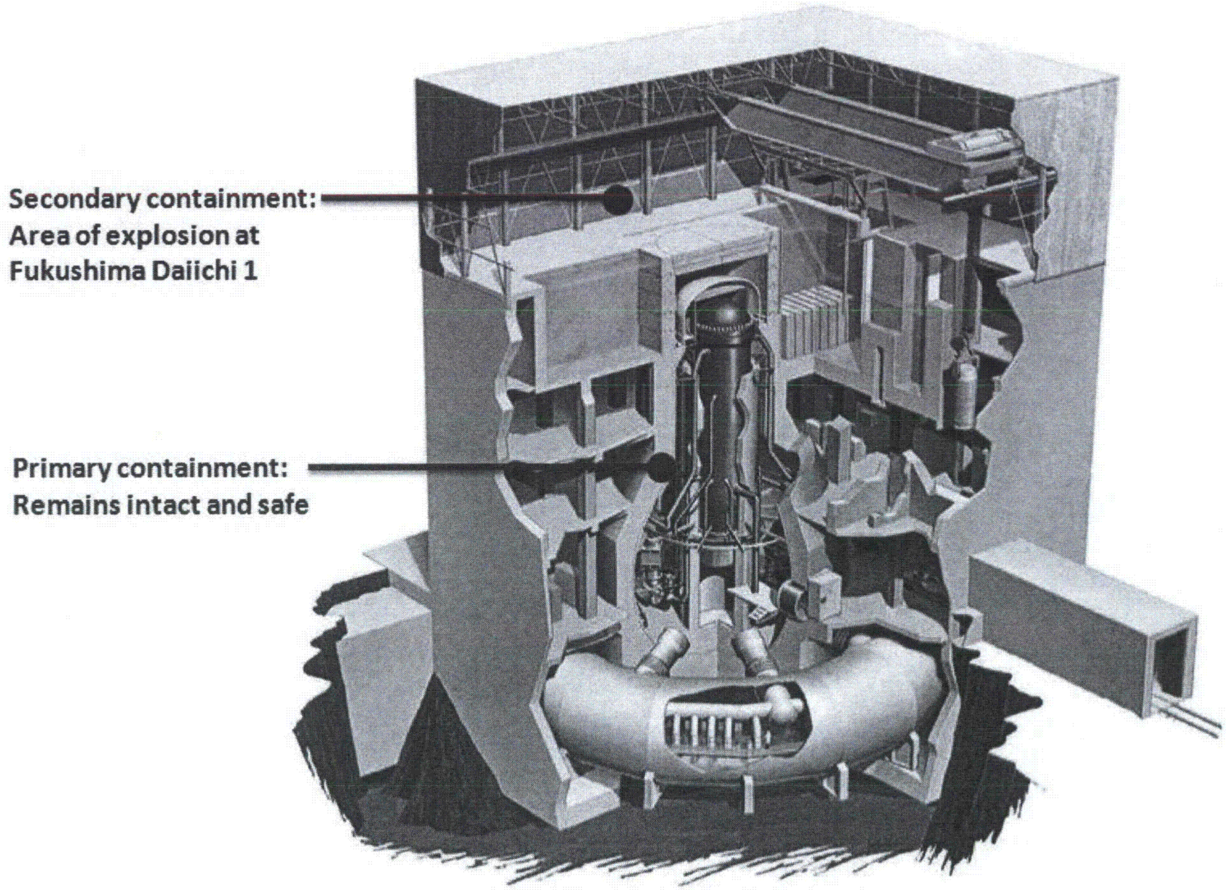
BWR with Mark 1 Containment



Boiling Water Reactor Design

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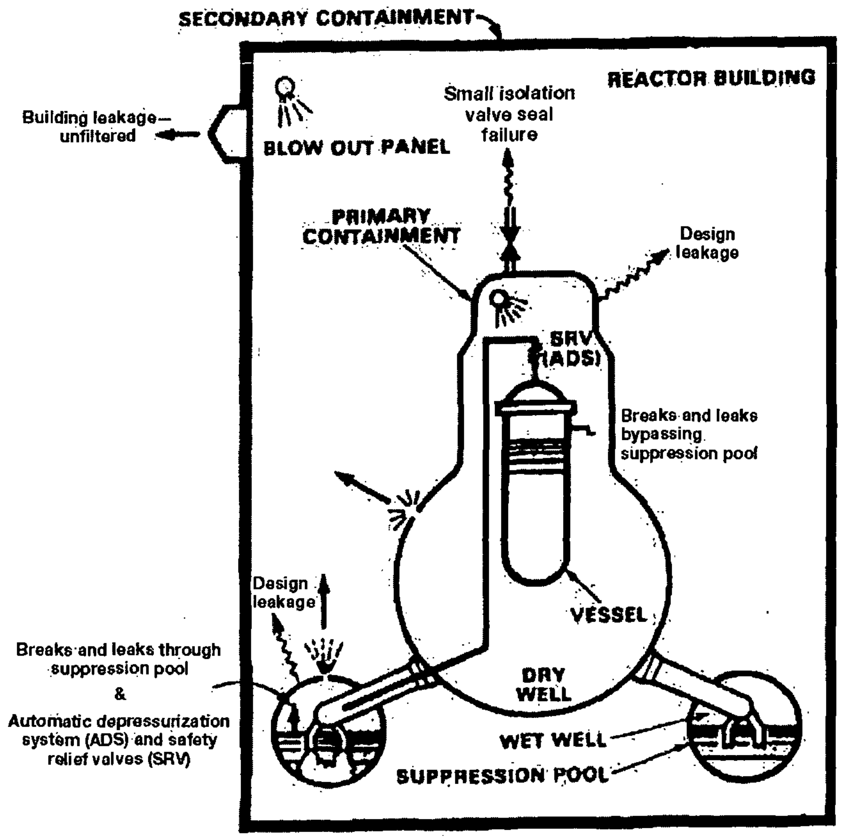
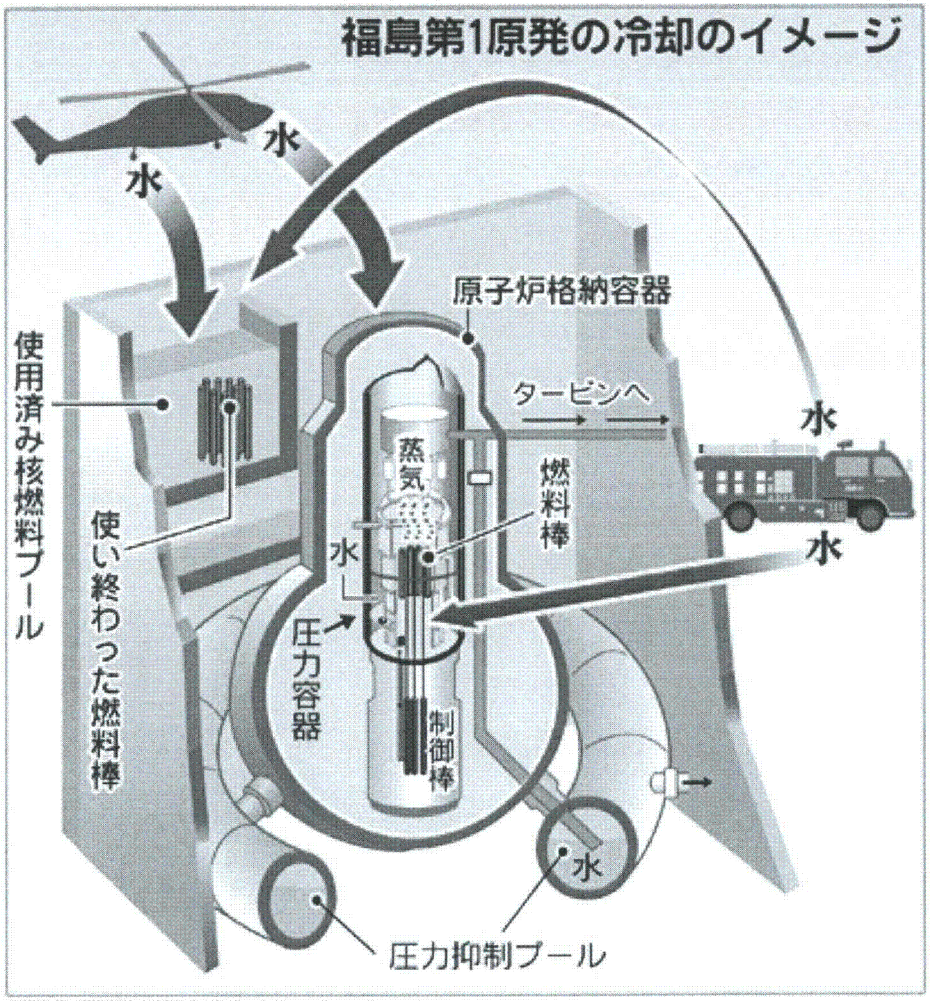
Mark I Containment Release Pathways Simplified



Most Recent View of Units 3 & 4

Japanese Depiction of Cooling Water Sources at Units 3 & 4 (Prior to the Return of Offsite Power)



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Current Conditions - NRC's Assessment

- Units 1, 2, 3 Stable w/some degree of core damage. Being cooled with fresh water.
- Units 2 and 3 some primary containment damage. Releases of radioactivity including significant contamination in the lower levels of the Unit 2 and Unit 3 turbine buildings.
- The spent fuel pools on Units 1-4 have experienced varying water levels, but also have been receiving seawater from helicopters and spray systems.

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Current Conditions - NRC's Assessment Cont.

- The U-2 spent fuel pool receiving fresh water and they are trying to change all the units from fire trucks to normal pumping in the next few days.
- Tokyo Electric Power Company has restored electric power to the site and the six reactor control rooms, and the situation, in general, continues to further stabilize, although many hurdles remain.

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NRC Response Efforts

- NRC continues to monitor the unfolding events in Japan.
- NRC is coordinating their response with other federal agencies.
- NRC has deployed a team to Tokyo.
- NRC providing technical assistance to the U.S. Ambassador in Japan and the Japanese Government.
- NRC continues assessment of radiological conditions, dose projections, and protective action recommendations.
- NRC Chairman Jaczko in Japan this week and keeps White House apprised.

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Ensuring Reactor Safety

- General Design Criteria (10CFR50, Appendix A) lay out the deterministic basis for the design of nuclear power plant safety systems.

- In 1975 NRC completed its first PRA study and continues to evaluate the risks to the public from the operation of nuclear power plants to within our safety goals by limiting the chance of core damage and fission product release to the environment.

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Ensuring Reactor Safety

- Significant activity to evaluate the chance and consequences of a Station Blackout (SBO Rule 10CFR50.63 1988) plant procedures and changes implemented in the 1990s.

- Generic Letters 88-20 “Individual Plant Examination for Severe Accident Vulnerabilities”

- NRC Maintenance Rule (10CFR50.65, 1991) Implemented in 1996

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Ensuring Reactor Safety

- In 2000 the NRC implemented the Reactor Oversight Program (ROP).
- Following September 11, 2001, the NRC and industry conducted detailed assessments . NRC issued orders for licensees to take actions to develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following areas: (i) Fire fighting; (ii) Operations to mitigate fuel damage; and (iii) Actions to minimize radiological release.

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NRC Initiatives

- NRC Issued Information Notice 2011-005
- NRC Commission supported the establishment of an agency task force.
- Temporary Instruction 2515/183
- Ongoing Communications with the public, Congressional, State (SLO), Local Agencies

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Overview of Generic Issue 199

Updated Seismic Hazard Estimates

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Background: GI-199 Safety/Risk Assessment Context and Results

- Generic Issues Program Stages
  - Identification – Early Site Permit reviews
  - Acceptance
  - Screening
  - Safety/Risk Assessment
  - Regulatory Assessment
- Safety / Risk Assessment Results
  - Operating power plants are safe.
  - Overall seismic risk estimates remain small
  - The new seismic data for some plants meet the criteria for further evaluations

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GI-199 Safety / Risk Assessment Assumptions

- Performed a conservative, screening-level assessment to evaluate whether further investigations are warranted.
  - The nature of the information used (seismic hazard data, plant-level fragility information) make these estimates useful only as a screening tool.
  - The results should not be interpreted as definitive estimates of plant-specific seismic risk because some analyses were conservative making the calculated risk higher than in reality.

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GI-199 Current Status

- Evaluating plant-specific information to determine if improvements to seismic safety are warranted

- Additional information is needed to consider plant-specific backfits

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Next Steps for GI-199

- Issued an Information Notice to inform plants of the GI-199 Safety/Risk Assessment results. (September 2010)

- NRC is developing a generic communication to request needed data. (2011)

Lee, Richard

From: Lee, Richard
Sent: Wednesday, March 30, 2011 2:05 PM
To: Salay, Michael; 'Michael Salay'
Subject: Your names on the list to go to Japan

Importance: High

Mike:

Your name is among the 4 that was sent to the Chairman for approval to go to Japan. We should hear back by this afternoon or late tonight. Chairman up on the Hill this morning.

If you go, you can leave on Sunday. You are to replace one who will be returning to U.S. on 4/06 or 4/07.

Richard

4/29/6

Lee, Richard

From: Lee, Richard
Sent: Wednesday, March 30, 2011 3:08 PM
To: Case, Michael
Cc: Sheron, Brian; Uhle, Jennifer; Gibson, Kathy
Subject: RE: 3rd Team to Japan

Thanks, Mike:
I spoke to Michele Evans earlier.
Richard

-----Original Message-----
From: Case, Michael
Sent: Wednesday, March 30, 2011 3:05 PM
To: Gibson, Kathy
Cc: Sheron, Brian; Uhle, Jennifer; Lee, Richard
Subject: 3rd Team to Japan

Hi Kathy

Just a quick update from Michele. She is still waiting for feedback from the Chairman on the size of the team but it looks like Mike Salay is still on the short list.

Michele has been in contact with Richard and as soon as she gets the OK she'll let Richard know so he can get Mike back from Europe.
Sent from Blackberry
Michael Case.

4/297

A11

Lee, Richard

From:	Richard L Garwin [rlg2@us.ibm.com]
Sent:	Wednesday, March 30, 2011 5:59 PM
To:	Binkley, Steve
Cc:	Brinkman, Bill; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Harold McFarlane; Harold Denton; Adams, Ian; John Holdren; JOE H. PAYER; Kelly, John E (NE); John Grossenbacher; Owens, Missy; Per Peterson; Lyons, Peter; Phil Finck; Dick Garwin; Lee, Richard; Bob Budnitz; Rolando Szilard; SCHU; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject:	Measuring water level in dry well by coupling to the organ-pipe resonance of the contained air?

I'll estimate this.

Dick Garwin

4/298

Lee, Richard

From: Lee, Richard
Sent: Thursday, March 31, 2011 7:49 AM
To: 'Gauntt, Randall O'
Subject: RE: Mike Salay is on his way to Japan soon

Great. Where are you working out of?

-----Original Message-----
From: Gauntt, Randall O [mailto:rogaunt@sandia.gov]
Sent: Thursday, March 31, 2011 2:55 AM
To: Lee, Richard
Subject: Re: Mike Salay is on his way to Japan soon

OK. We have arrived on Thursday PM.
I am here with Jeff LaChance. We are expecting to be here 2 weeks minimum. Who knows.
Randy

----- Original Message -----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Wednesday, March 30, 2011 08:43 PM
To: Gauntt, Randall O
Subject: Mike Salay is on his way to Japan soon

Randy:

Mike is leaving for Japan on Sunday, April 3. I asked him to return from the Phebus meeting in the Netherlands.

Richard

U/299

Beasley, Benjamin

From: Boska, John
Sent: Thursday, March 31, 2011 8:56 AM
To: Kauffman, John
Cc: Beasley, Benjamin; Khanna, Meena; Jessup, William; Salgado, Nancy
Subject: RE: RE: Outcomes from Meeting With New York State Officials
Attachments: image001.gif

Just to inform everyone on the use of 10CFR50, Appendix A, General Design Criteria, I refer to this wording we use in Indian Point license amendments:

"The following explains the applicability of General Design Criteria (GDC) for IP2 and IP3. The construction permits for IP2 and IP3 were issued by the Atomic Energy Commission (AEC) on October 14, 1966 and August 13, 1969, and the operating licenses were issued on September 28, 1973, and December 12, 1975. The plant GDC are discussed in the Updated Final Safety Analysis Report (UFSAR) Chapter 1.3, "General Design Criteria," with more details given in the applicable UFSAR sections. The AEC published the final rule that added Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for IP2 and IP3 are those in the UFSARs."

This same information applies to many older reactors. In all cases, we should just refer to the UFSAR. I will edit this reply to delete references to GDC 2, and just reference the UFSAR (the effect in this case is the same). As a general principle, appendices to 10CFR50 do not apply to plants unless there are specific words invoking the appendix. For example, here are quotes from Appendix A:
"Under the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility."
"Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified."

So the only way to tell which 10CFR50 Appendix A GDC apply to a plant is to read the UFSAR. They don't apply in a blanket manner.

John Boska
Indian Point Project Manager, NRR/DORL
U.S. Nuclear Regulatory Commission
301-415-2901
email: john.boska@nrc.gov

From: Kauffman, John
Sent: Thursday, March 31, 2011 7:17 AM
To: Boska, John
Cc: Beasley, Benjamin
Subject: RE: Outcomes from Meeting With New York State Officials

John,
Please see my input below on follow-up items assigned to me. Please contact me if you have any questions or need further assistance.

4/300

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

John V. Kauffman
Senior Reactor Systems Engineer
US NRC/RES/DRA/OEGIB
Washington, DC 20555 Mail Stop: C-2A07M
Phone: 301-251-7465
Fax: 301-251-7410

Please visit the internal GIP web page or external GIP web page.

1) GI-199 Safety/Risk Assessment Report

Ben Beasley previously provided you the link to the GI-199 Safety/Risk Assessment report, including the memorandum and all enclosures/attachments. Ben also provided electronic versions of the following publically available documents:

ML100270598 (Transmittal Memo)
ML100270639 (Safety/Risk Assessment (S/RA) report)
ML100270064 (Appendix A of the S/RA report)
ML100270691 (Appendix B of the S/RA report)
ML100270731 (Appendix C of the S/RA report)
ML100270756 (Appendix D of the S/RA report)

2a) Discussion of Spent Fuel Pools and GI-199 (this is from a write-up provided by Meena Khanna that Billy Jessup put together)(please note that I have 1 question/suggested edit in the next to last paragraph)

Spent fuel pools (SFPs) were not specifically evaluated as part of GI-199. However, based on their design attributes (as follows), SFPs remain safe. SFPs are constructed of reinforced concrete, several feet thick, with a stainless steel liner to prevent leakage and maintain water quality. Due to their configuration, SFPs are inherently structurally-rugged and are designed to the same seismic requirements as the nuclear plant. Information Notices related to GI-199 were sent to nuclear power plant licensees and licensees of independent spent fuel storage installations (ISFSIs).

Note: Typically, SFPs are about 40 feet deep and vary in width and length. The fuel is stored in stainless steel racks and submerged with approximately 23 feet of water above the top of the stored fuel. Each plant has a preferred SFP make-up water source (the refueling water storage tank for pressurized water reactors and the condensate storage tank for boiling water reactors). SFPs have alternate means of make-up such as service water systems and the fire water system. SFPs are also typically designed (e.g. with anti-siphon check valves) and instrumented such that leakage is minimized and promptly detected.

There has been a previous Generic Issue (GI-173) concerning spent fuel pool safety (ML013520142). This issue was closed in 2001 with no new requirements. In resolving this issue, the staff implemented an action plan for operating reactors that involved: gathering technical information for all operating reactors through plant visits, reviews of design and licensing documents, and performance of a survey using regional personnel; analyzing capabilities to maintain safe storage conditions for irradiated fuel at each site; and developing proposed actions to address identified concerns. For representative plants having one or more of the design features of concern, the staff estimated the frequency of a significant loss of coolant inventory or a sustained loss of cooling, which were selected as conservative surrogate conditions for fuel damage. These estimated frequencies were compared against screening criteria developed for reactor accidents to assess the need for new or revised requirements. The screening criteria used for comparison with the endstate frequencies were: below 1E-06 per year, no action; between 1E-06 and 1E-05 per year, engineering judgement was used to determine need for detailed evaluation; and above 1E-05, a detailed evaluation would be performed. Several licensees took voluntary actions to address the concerns identified at their facilities. For the remaining facilities, the staff concluded that no new or revised requirements were justified.

The Japanese earthquake did not change our understanding of the seismic hazard at U.S. nuclear power plants or the conclusions of GI-199. This is because the effect of a single earthquake is small on the estimated seismic hazard, unless it occurs in an area not previously recognized as being capable of producing earthquakes, or is larger than previously believed possible in a region. In a seismic hazard study, the seismic source zones are specifically delineated to include a sufficient number of earthquakes to provide a stable estimate of the seismicity rate and are thus relatively insensitive to the addition of a single earthquake. If an earthquake does occur in an area not previously recognized as being capable of producing earthquakes or if an earthquake occurs that is larger than previously believed possible in a region, changes to the seismic hazard model used to develop seismic hazard estimates would be required. This Japanese earthquake occurred on a "subduction zone", which is the type of tectonic region that produces earthquakes of the largest magnitude. A subduction zone is a tectonic plate boundary where one tectonic plate is pushed under another plate. Subduction zone earthquakes are also required to produce the kind of massive tsunami seen in Japan. In the continental US, the only subduction zone is the Cascadia subduction zone which lies off the coast of northern California, Oregon and Washington. So, a continental earthquake and tsunami as large as in Japan could only happen there. Nevertheless, the NRC intends to conduct an extensive lessons learned evaluation of the Japanese earthquake and tsunami. NRC will enhance our regulatory program as appropriate based on the results of the lessons learned evaluation.

2b) <u>Discussion of Indian Point Spent Fuel Pools</u>

General Information:

GDC 2 requires that structures important to safety be designed to withstand the effects of natural phenomena combined with those of normal and accident conditions without loss of capability to perform their safety function. As such, all structures at Indian Point Units 2 and 3 (IP2 and IP3, respectively), including the spent fuel pools (SFPs), which fall under this classification are designed to withstand loads due to earthquakes, in combination with other loads.

Load combinations and specifications cited in SRP Section 3.8.4, "Other Seismic Category I Structures," provide acceptable engineering criteria to accomplish that function for structures such as SFPs. Meeting these requirements provides added assurance that safety-related structures will be designed to withstand the effects of natural phenomena and will perform their intended safety function.

Indian Point Units 2 and 3:

Chapter 9 of the IP2 and IP3 Final Safety Analysis Reports (FSARs) indicates that the SFP structures are classified as Seismic Category I. The IP2 FSAR is specific regarding the design criteria, and indicates that the IP2 SFP was designed in accordance with the provisions of American Concrete Institute (ACI)-318, "Building Code Requirements for Reinforced Concrete" (see Section 9.5.2.1.4 of the IP2 FSAR). The 1989 license amendment issued for IP3 SFP re-rack indicates that the design criteria used to evaluate the SFP structure are based on the provisions in ACI 349-80, "Code Requirements for Nuclear Safety-Related Concrete Structures."

As indicated above, based the classification of these structures, they are required to be designed against bounding loading combinations which include loads due to a safe shutdown earthquake. As such, the structural analyses are performed to ensure that the SFPs will remain functional during and after a safe shutdown earthquake.

The following licensing actions relate to the structural analysis of the IP2 and IP3 SFPs for conditions which include loads due to seismic events:

The NRC issued a license amendment in 1989 for a re-rack of the IP3 SFP (ML003778816). An extensive review of the structural aspects of the re-rack was performed by Brookhaven National Laboratory (BNL) and is included as Appendix A of the NRC staff's safety evaluation associated with this amendment. This review explicitly notes that the licensee demonstrated that the design basis requirements associated with the IP3 SFP would continue to be satisfied following the re-rack. As such, the licensee demonstrated that under design basis loading combinations, which include loads due to a safe shutdown earthquake, the applicable ACI provisions would continue to be satisfied following the re-rack.

The NRC issued a license amendment in 1990 for a re-rack of the IP2 SFP (ML003778320). In the NRC staff's associated safety evaluation, it was noted that the licensee evaluated the effects of the high density racks on the SFP structure and concluded that the structure would continue to satisfy the design basis requirements prescribed by the ACI code. These design basis requirements include withstanding the loads generated under a safe shutdown earthquake.

Subsequent to the re-rack of the IP2 SFP, the NRC staff's review of the licensee's request to review [should this be renew?] the IP2 operating license included a number of audit items. Audit Item 360 associated with the renewal of the IP 2 operating license focused on the leakage previously discovered at the IP2 site. The licensee provided information to the NRC staff by letter dated November 6, 2008, which presented the results of structural evaluations (finite element analyses) performed for the IP2 SPF walls. The model used in this structural analysis accounted for bounding conditions which may exist due to the potential for degradation resulting from the SFP leakage (i.e., no credit for reinforcing steel). This is a very conservative assumption given that all testing performed by the licensee up to the date of the November 6, 2008, submittal demonstrated that there was no concern for degradation of the rebar due to boron concentrations resulting from SFP leakage. The results of the analysis showed that the structure contained significant margin against failure when the structure was subjected to design basis loading conditions, including those due to a design basis earthquake, even if no rebar was considered in the model.

The overall conclusion which is demonstrated by the re-rack evaluations and the additional IP2 SFP evaluation is that the licensee has shown multiple times that the design basis requirements associated with the design of the SFP structure are satisfied. As such, the licensee has demonstrated that there is reasonable assurance that the structure will maintain its ability to serve its safety function during and after a safe shutdown earthquake or other natural phenomena.

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Thursday, March 31, 2011 11:38 AM
To: Lee, Richard; Kelly, John E (NE)
Subject: Test CST with Seawater

John, I spoke to Nenoff. She has a commercial sample of the crystalline silicon titanates and thinks she can test with seawater if you don't have someone already positioned to do the testing. Dana

4/301

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Thursday, March 31, 2011 1:12 PM
To: Ibarra, Jose
Subject: RE: DRA Support to Japanese Event

In conjunction with NRR and Region 1, Doug Coe, Marty Stutzke and Ben Beasley have supported meetings with the Governor's offices from New York and Massachusetts.

From: Ibarra, Jose
Sent: Thursday, March 31, 2011 12:51 PM
To: Barnes, Valerie; Nicholson, Thomas; Siu, Nathan; Stutzke, Martin; Ott, William; Salley, MarkHenry; Peters, Sean; Beasley, Benjamin; Coyne, Kevin; Demoss, Gary
Subject: DRA Support to Japanese Event

All,
DRA would like to take credit for special support given related to the Japanese Event. Can you in one sentence or two tell me what special support you have provided. I will construct the text for an DRA accomplishment to be included in the OP Plan update. I need that information today to meet the due date of the Op Plan update. Thanks. Jose

41302

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Thursday, March 31, 2011 4:35 PM
To: Adams, Ian
Cc: Brinkman, Bill; Narendra, Blake; Hurlbut, Brandon; Sheron, Brian; Butnitz, Bob (pacbell.net); Smith, Haley; McFarlane, Harold; Adams, Ian; Kelly, John E (NE); Grossenbacher, John (INL); Pitzer, Karrie S.; Chambers, Megan (S4); Owens, Missy; Miller, Neile; Fitzgerald, Paige; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Steve Fetter; Aoki, Steven; Binkley, Steve; Mustin, Tracy
Subject: Useful website for technical details vs time.

http://www.nisa.meti.go.jp/english/files/en20110331-2-2.pdf

More generally, http://www.nisa.meti.go.jp/english
Don't be put off by the titles of the press releases.

Dick Garwin

W303

Bensi, Michelle

From: Bensi, Michelle
Sent: Thursday, March 31, 2011 4:53 PM
To: Kauffman, John
Subject: RE: Reminder--OEGIB Weekly Activities Input due by noon tomorrow, Friday 4/1/2011. [eom]

Thanks,
Shelby

Last week activities

- Seismic Q&A document in response to events in Japan
- Presentation (and prep) for joint branch meeting

Next week activities

- Seismic Q&A document
- Out-of-office Friday (CHU)

From: Kauffman, John
Sent: Thursday, March 31, 2011 3:59 PM
To: Bensi, Michelle; Criscione, Lawrence; Ibarra, Jose; Killian, Lauren; Lane, John; Reisifard, Mehdi; Perkins, Richard; Salomon, Arthur; Smith, April; Wegner, Mary
Subject: Reminder--OEGIB Weekly Activities Input due by noon tomorrow, Friday 4/1/2011. [eom]

4/304

Bonaccorso, Amy

From: CL Spriggs [boardwaxmax@excite.com]
Sent: Friday, April 01, 2011 12:09 PM
To: NRC Allegation
Subject: Website issue

Hello:

Please check this website.

http://en.wikipedia.org/wiki/List_of_nuclear_reactors

If you scroll to the bottom, the U.S. nuclear plants in the NE are listed with GPS coordinates.
Realize this is not your site, but damn, how smart is that?

Cheers
Craig

4/305

Beasley, Benjamin

From: Lane, John
Sent: Friday, April 01, 2011 10:06 AM
To: Hogan, Rosemary; Stutzke, Martin; Perkins, Richard; Bensi, Michelle
Cc: Beasley, Benjamin; Kauffman, John
Subject: Seismic Review Table
Attachments: Seismic Review Table_ML1108807472.pdf

Attached is an old NUREG/CR that I worked on back in 1980 entitled, the Seismic Review Table. With the exception of myself, everyone else involved with it is either retired, deceased or both (a causal relationship has not been established between the report and those ends).

It's a tabulation of the FSAR approved seismic/structural designs for the plants that were in house as of that timeframe (which is most of the current fleet). It includes a relatively comprehensive view of each plant's design in terms of OBE/SSE earthquake level and spectra, the soil-structure assumptions, and the containment design loads. It even includes proximity to local dams.

It is in ADAMS and I'm considering making it publicly available at some point. (If you have any opinion on that, pls. let me know.)

I hope it's of some value to you in either the Japan-related, GI-199, or pre-GI dam-related on-going efforts.

jcl

U/306

NUREG/CR-1429

Seismic Review Table

Prepared by M. Subudhi, M. Reich, B. Koplik, J. Lane

Department of Nuclear Energy
Brookhaven National Laboratory

Prepared for
U. S. Nuclear Regulatory
Commission

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-1429

Seismic Review Table

Manuscript Completed: April 1980
Date Published: May 1980

Prepared by
M. Subudhi, M. Reich, B. Koplik, J. Lane*

Structural Analysis Group
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, NY 11973

*Staff, U.S. Nuclear Regulatory Commission

Prepared for
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A3326

ABSTRACT

The Seismic Review Table is a summary of Engineering Design parameters that were employed in the seismic analysis and design of nuclear power plants. The table covers 71 reactors licensed to operate by the U.S.N.R.C. The information contained is listed plant by plant and consists of OBE and SSE "g" Level and Modified Mercalli Intensity; Earthquake Time History used to develop the ground response spectra or as input in the dynamic analysis; Number of Earthquake Components used and Method of Combining Them; Method of Modal Combination; Type of Ground Design Spectra; Method of Generation of Floor Response Spectra; Type of Foundation and Depth; Type, Thickness, Shear Wave Velocity and Shear Modulus Profile of the Surrounding Subgrade Soil and Bedrock; Ground Water Table Depth; nearby Dams; Modelling Method used for soil-structure interaction; Material Damping of Soil; Limitation on Modal Damping . Damping Values; and Loading Combinations, and Acceptance Criteria for Category I Structures, Mechanical Equipment, Piping, and Electrical systems. The goal of the Seismic Review Table is to provide a reference of the available information relevant to the seismic design of currently licensed nuclear power plants.

TABLE OF CONTENTS

ACKNOWLEDGEMENTS

The authors wish to acknowledge their indebtedness and gratitude to various people consulted during the preparation of the Seismic Review Tables. Particular thanks are due to our Department Librarians, Mrs. Helen Todosow and Mrs. Catherine Green for their help in gathering and obtaining the various FSAR's, amendments, etc. and to Dr. C. P. Tan of the Structural Engineering Branch, NRC, for the Containment Vessel data shown in Table I of this report. Grateful acknowledgement is also due to Larry Shao, Acting Assistant Director of Engineering Programs, Division of Operating Reactors and Assistant Director for General Reactor Safety Research, NRC, and Dr. P. T. Kuo, Section Leader, Seismic Review Group, NRC, for their constructive criticism and advice regarding the contents of the review tables. Finally to Miss Joan Murray who with patience typed and retyped the corrected drafts, our sincerest gratitude is due.

INTRODUCTION

The intent of this report is to enable a quick reference of the major seismic design parameters inherent in the 71 currently licensed nuclear power plants. All of the presented data was obtained from the existing Final Safety Analysis Reports (FSAR) and their associated amendments. The results are tabulated for each plant in a five page "Seismic Review Table." The major headings in the table are:

- A) Earthquake data
- B) Method of combination (e.g., modes and earthquakes directional components
- C) Design spectra
- D) Foundation and liquefaction assessment
- E) Soil-structure interaction
- F) Damping, load combination and acceptance criteria and allowable stresses for:
    - 1) Category I structures
    - 2) Mechanical Equipment and piping
    - 3) Electrical equipment

Table I lists all of the plants together with the names of the owners, the location, the principal reactor contractor, the plant architectural engineers, the type of plant (PWR, BWR, HTGR), the type of containment vessel, and the electrical and thermal power output. FSAR's for all the plants listed in the table have been reviewed and the tabulated results are given in this report. For completeness Figure 1 depicting the geographical locations of the operational plants is also included.

PROGRAM TASKS AND ACCOMPLISHMENTS

Efforts under this program can be subdivided into three distinct stages: Stage 1 involved the determination and collection of all available plant FSAR's and related questions, answers, and amendments. Next, under Stage 2, the collected information was reviewed in detail for relevance to the information needed for the Seismic Review Table. Finally, under Stage 3, the pertinent parameters were assembled and summarized in tabular form.

With reference to the work carried out under Stage 1, it should be realized that the documented information contains numerous sections, subsections, and amendments per plant which were compiled over a span of many years. This information had to be reviewed to ascertain which documents were available and which had to be ordered. This was accomplished by carrying out a careful review of the documents and comparing the information contained within the documents against the information compiled in the following reference reports:

Title Listing of Civilian Power Reactor Docket Literature in Nuclear Science Abstracts, volumes 21-26 (1967-1972), TID-3354 R1. U.S. Atomic Energy Commission, Technical Information Center, April 1973.

Title Listing of Civilian Power Reactor Docket Literature in Nuclear Science Abstracts, volumes 27 (Jan.-June 1973), TID-3324-R1-S1. U.S. Atomic Energy Commission, Technical Information Center, September 1973.

Title Listing of Power Reactor Docket Information, PRDI-74-12. U.S. Atomic Energy Commission, Technical Information Center, December 1974.

Power Reactor Docket Information, Annual Cumulation, NUREG/PRDI-75/12. U.S. Energy Research and Development Administration, Technical Information Center, December 1975.

Power Reactor Docket Information, Annual Cumulation, NUREG/PRDI-76/12/P1. U.S. Energy Research and Development Administration, Technical Information Center, December 1976.

Power Reactor Docket Information, Annual Cumulation, NUREG/PRDI-77/12/P1. U.S. Dept. of Energy, Technical Information Center, December 1977.

Power Reactor Docket Information, Annual Cumulation, NUREG/PRDI-78/12/P1. U.S. Dept. of Energy, Technical Information Center, December 1978.

Since there was no specific standardized FSAR format until 1975-76, each FSAR had to be examined on an individual basis. In a number of cases the FSAR was actually defined as an amendment to the PSAR. Once it was determined what information was missing and what part of the missing information involved seismic design criteria, the necessary steps were taken to obtain the required documents.

Once the material needed for the review was compiled, Stage 2 efforts were initiated. For each plant assembled FSAR's were first reviewed for the pertinent seismic information. These were available either in "hard cover" or in "microfiche" form. Next, the amendments which include various questions and answers about the plant raised over a period of many years were reviewed and the gathered information was then compiled and referenced for section and page number.

Under Stage 3, the compiled reference material of Stage 2 was prepared and extracted for insertion into the Seismic Review Tables. The information given in the table thus reflects the data up to an including the latest amendments available at time of publication. The tables are numbered according to the numbering scheme shown in the first column of Table I. For each number, a set of five pages comprising the Seismic Review Table is presented with the page number appearing in the lower right hand corner in sequence. As an example, page 8-2 would indicate the eighth entry on Table I, with the number 2 representing the second page of the five-page review table.

Referring to the Seismic Review Tables, the first item assembled is on page 1 of the five-page table. The name of the plant with reactor unit numbers (if more than one), the type of reactors, and containment, Nuclear Steam System Supplier (NSSS), the architect engineer, and the CP/OL issue dates. Next, under the heading of earthquake data, information pertaining to OBE, SSE, and earthquake time-history was assembled. The OBE and SSE information was further broken down into horizontal and vertical "g" values and Modified Mercalli Intensity values. Reference pages, sections, and amendment numbers are listed in the tables for all assembled information. Under the time history column, names of the earthquake records used are given. These records in turn are used either for the development of the ground design spectra or are modified so that their response spectra envelopes the specified ground design spectra. Generally speaking, this information was available for most of the plants. However, some of the early plants, such as Yankee Rowe, did not have this information in the reviewed dockets, and thus the term "not available" is written in the table. For those cases where the available information was unclear, the term "unclear information" appears in the table, together with the pertinent page numbers where the unclear information is given so that the reader can look up the information for further insight.

Returning to headings OBE and SSE, in many plants the vertical components were equal to two-thirds of the horizontal, with OBE values typically one-half of the SSE. For the earthquake time-history, the older plants usually used El Centro or Taft, while the newer plants used synthetic time-histories.

Methods of combinations were assembled under the subheadings "Number of Earthquake Components Used and Its Combination" and "Modal Combination." The information under these headings includes such items as the the number of horizontal and vertical components used for the analysis, the number of modes considered, and how they were combined, e.g., absolute sum, SRSS, or algebraic sum. It is to be noted that the term "modal combination used" in the table refers to the response spectrum analysis.

The final item on page 1 involves the design spectra with the two subheadings entitled "Type of Ground Design Spectra" and "Method of Generation of Floor Response Spectra." Ground design spectra includes the Housner, Newmark, and Regulatory Guide 1.60 response spectra or any other method specified in the FSAR's. The most commonly used method for generating the floor response spectra was the time-history method. When information regarding the input time-history was available, it was also included under this heading. For some of the older plants, the ground design spectra was directly used with some amplification factor.

Turning to page 2 of 5 of the table, the major headings are "Foundation and Liquefaction Assessment" and "Soil-Structure Interaction." The first item contains four subtopics: "Type of Foundation," "Bearing Information" (including information related to the type, thickness, and shear velocity profile), "Groundwater Table," and "Dams." Foundation description and bedrock characteristics are listed for the containment building. Information regarding structures on pile foundations is also given under this heading. Bearing Information lists such items as type of rock (dolomite, glacial fill, sandstone, etc.), the thickness of the various soil deposits, and shear wave velocities. Groundwater Table information and the existence of nearby dam locations were obtained from the site geological survey.

"Soil Structure Interaction" consists of four subtopics. "Method of Modelling" lists the mathematical model chosen for generating the floor response spectra of the reactor building and the soil beneath it. Usually the structure is modeled as a conventional stick model while the soil is represented as either a lumped spring or finite element model. It is to be noted that a number of plants have their foundation on bedrock. When reviewing the soil structure interaction modelling method, it was found that for some plants a fixed base method was employed. For these cases, the notation fixed base method appears. For cases where no statement was found as to the type of modelling used, the term "not available" was entered in the table. The term "not available" should only be interpreted as a statement of fact with reference to the material presented in the FSAR; it only means that no information about the particular item was found. Other subtopics include the "Soil Shear Strength Modulus Profile," "Material Damping of Soil," and the "Limitation on Modal Damping."

Pages 3, 4, and 5 of the Seismic Review Table are devoted respectively to Category I--structure, mechanical, piping and electrical equipment. Each of these pages have common headings that include "Damping Values" (OBE/SSE) and "Design Criteria," with the latter heading containing subheadings for load combination and acceptance criteria/allowable stresses. "Method of Qualification" (testing or analytical) was included for the mechanical equipment, piping and electrical equipment given on pages 4 and 5. Generally, very little information was available for electrical equipment.

The information listed for the 11 SEP plants (Big Rock Point, Dresden 1 and 2, Ginna, Haddam Neck, LaCrosse, Millstone 1, Oyster Creek, Palisades, San Onofre 1, and Yankee Rowe) was partly obtained through the use of unpublished docket search reports supplied to us by the Systematic Evaluation Program Branch, DOR. This information supplements what was obtained by Brookhaven staff members in their docket search.

In conclusion, this report contains much information covering a wide range of seismic topics. It is possible that some relevant information has been inadvertently overlooked. The Structural Engineering Branch of the Division of Engineering has the responsibility for maintaining these tables and would appreciate any contribution from interested parties as to additions or modifications which might be made to improve it.

The information contained here comprises a data base which will be used to evaluate conformance of the operating reactors with current seismic design guidelines.

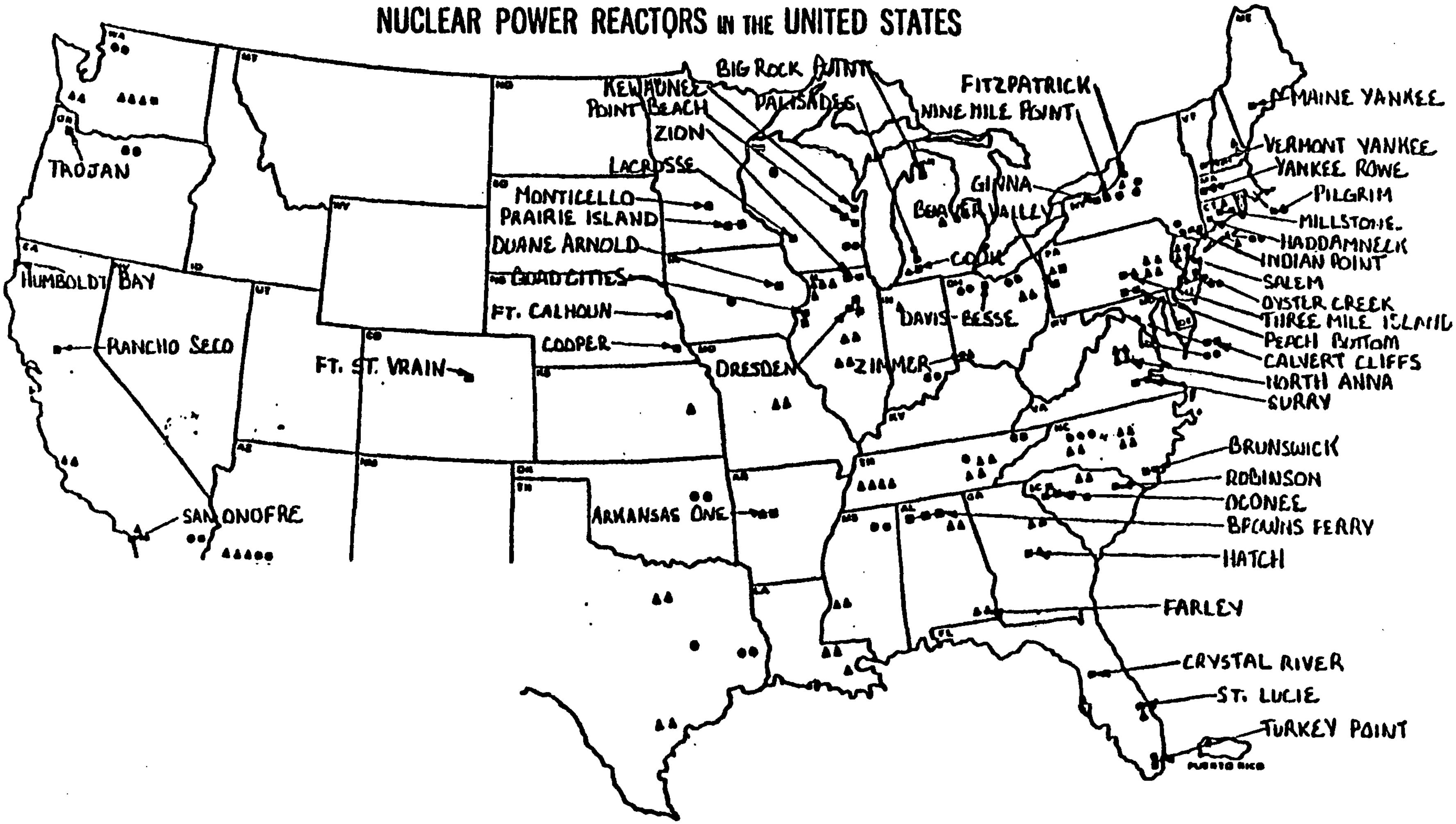
NUCLEAR POWER REACTORS IN THE UNITED STATES
TROJAN
HUMBOLDT BAY
RANCHO SECO
SAN ONOFRE
FT. ST. VRAIN
MONTICELLO
PRAIRIE ISLAND
DUANE ARNOLD
QUAD CITIES
FT. CALHOUN
COOPER
ARKANSAS ONE
KEWAUNEE
POINT BEACH
ZION
LACROSSE
BIG ROCK POINT
PALISADES
DRESDEN
ZIMMER
COOK
DAVIS-BESSE
BEAVER VALLEY
GINNA
FITZPATRICK
NINE MILE POINT
MAINE YANKEE
VERMONT YANKEE
YANKEE ROWE
PILGRIM
MILLSTONE
HADDAMNECK
INDIAN POINT
SALEM
OYSTER CREEK
THREE MILE ISLAND
PEACH BOTTOM
CALVERT CLIFFS
NORTH ANNA
SURRY
BRUNSWICK
ROBINSON
OCONEE
BROWNS FERRY
HATCH
FARLEY
CRYSTAL RIVER
ST. LUCIE
TURKEY POINT
PUERTO RICO

FIGURE 1

CONTENTS

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufac-turer **	Architect Engineer **	Reac-tor Type	Containment Type *	Power	
							Unit Size Net MW(e)	Reactor MW (t)
1-1	Arkansas Nuclear One, Unit 1 (Arkansas Power & Light Co.)	Russellville, Ark.	B&W	Bechtel	PWR	(11)	850	2,568
2-1	Arkansas Nuclear One, Unit 2 (Arkansas Power & Light Co.)	Russellville, Ark.	Comb.	Bechtel	PWR	(11)	912	2,815
3-1	Beaver Valley Power Station, Unit 1 (Duquensne Light Co., Ohio Edison Co., and Pennsylvaina Power Co.)	Shippingport, Pa.	West.	S&W	PWR	(7)	852	2,652
4-1	Big Rock Point Plant Nuclear (Consumer Power Co.)	Big Rock Point, Mich.	GE	Bechtel	BWR	(1)	72	240
5-1	Browns Ferry Nuclear Power Station, Unit 1 (Tennessee Valley Authority)	Decatur, Ala.	GE	TVA	BWR	(2)	1,065	3,293
[illegible]-1	Browns Ferry Nuclear Power Station, Unit 2 (Tennessee Valley Authority)	Decatur, Ala.	GE	TVA	BWR	(2)	1,065	3,293
[illegible]-1	Browns Ferry Nuclear Power Station, Unit 3 (Tennessee Valley Authority)	Decatur, Ala.	GE	TVA	BWR	(2)	1,065	3,293
6-1	Brunswick Steam Electric Plant, Unit 1 (Carolina Power & Light Co.)	Southport, N.C.	GE	UE&C	BWR	(5)	821	2,436
6-1	Brunswick Steam Electric Plant, Unit 2 (Carolina Power & Light Co.)	Southport N.C.	GE	UE&C	BWR	(5)	821	2,436
7-1	Calvert Cliffs Nuclear Power Plant, Unit 1 (Baltimore Gas & Electric Co.)	Lusby, Md.	Comb.	Bechtel	PWR	(10)	845	2,700
7-1	Calvert Cliffs Nuclear Power Plant, Unit 2 (Baltimore Gas & Electric Co.)	Lusby, Md.	Comb.	Bechtel	PWR	(10)	845	2,700
8-1	Cooper Nuclear Station (Nebraska Public Power District and Iowa Power and Light Co.)	Brownville, Nebr.	GE	B&R	BWR	(2)	778	2,381
9-1	Crystal River Nuclear Plant, Unit 3 (Florida Power Corp.)	Red Level, Fla.	B&W	Gilbert	PWR	(10)	825	2,452

TABLE I: CURRENTLY LICENSED REACTORS IN UNITED STATES

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufacturer **	Architect Engineer **	Reactor Type	Containment Type *	Power Unit Size Net MW(e)	Power Reactor MW (t)
10-1	Davis-Besse Nuclear Power Station, Unit 1 Cleveland Electric Illuminating Co.)	Oak Harbor, Ohio	B&W	Bechtel	PWR	(4)	906	2,772
11-1	Donald C. Cook Nuclear Power Plant, Unit 1 (Indiana and Michigan Electric Co.)	Bridgman, Mich.	West.	AEP	PWR	(6)	1,054	3,250
11-1	Donald C. Cook Nuclear Power Plant, Unit 2 (Indiana and Michigan Electric Co.)	Bridgman, Mich.	West.	AEP	PWR	(6)	1,100	3,391
12-1	Dresden Nuclear Power Station, Unit 1 (Commonwealth Edison Co.)	Morris, Ill.	GE	Bechtel	BWR	(1)	200	700
13-1	Dresden Nuclear Power Station, Unit 2 (Commonwealth Edison Co.)	Morris, Ill.	GE	S&L	BWR	(2)	794	2,527
13-1	Dresden Nuclear Power Station, Unit 3 (Commonwealth Edison Co.)	Morris, Ill.	GE	S&L	BWR	(2)	794	2,527
14-1	Duane Arnold Energy Center, Unit 1 (Iowa Electric Light & Power Co., Central Iowa Power Cooperative, and Corn Belt Power Cooperative)	Palo, Iowa	GE	Bechtel	BWR	(2)	538	1,593
15-1	Edwin I. Hatch Nuclear Plant, Unit 1 (Georgia Power Co.)	Baxley, Ga.	GE	Bechtel	BWR	(2)	786	2,436
16-1	Edwin I. Hatch Nuclear Plant, Unit 2 (Georgia Power Co.)	Baxley, Ga.	GE	Bechtel	BWR	(2)	795	2,436
17-1	Fort Calhoun Station, Unit 1 (Omaha Public Power District)	Fort Calhoun, Nebr.	Comb.	G&H	PWR	(9)	457	1,420
18-1	Fort St. Vrain Nuclear Generating Station (Public Service Co. of Colorado)	Platteville, Colo.	GA	S&L	HTGR	(9)	330	842
19-1	Haddam Neck Plant (Connecticut Yankee Atomic Power Co.)	Haddam Neck, Conn.	West.	S&W	PWR	(8)	575	1,825
20-1	H. B. Robinson Plant, Unit 2 (Carolina Power & Light Co.)	Hartsville, S. C.	West.	Ebasco	PWR	(9)	700	2,200

CURRENTLY LICENSED REACTORS IN UNITED STATES (continued)

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufac-turer **	Architect Engineer **	Reactor Type	Containment Type *	Power Unit Size Net MW(e)	Power Reactor MW (t)
21-1	Humboldt Bay Power Plant, Unit 3 (Pacific Gas & Electric Co.	Eureka, Calif.	GE	Bechtel	BWR	(1)	63	242
22-1	Indian Point Station, Unit 1 (Consolidated Edison Co. of New York, Inc.)	Buchanan, N.Y.	B&W	UE&C	PWR	(3)	265	615
23-1	Indian Point Station, Unit 2 (Consolidated Edison Co. of New York, Inc.)	Buchanan, N.Y.	West.	UE&C	PWR	(8)	873	2,758
24-1	Indian Point Station, Unit 3 (Power Authority of New York)	Buchanan, N.Y.	West.	UE&C	PWR	(8)	965	2,760
25-1	James A. FitzPatrick Nuclear Power Plant (Power Authority of the State of New York)	Scriba, N.Y.	GE	S&W	BWR	(2)	821	2,436
26-1	Joseph M. Farley Nuclear Plant, Unit 1,2 (Alabama Power Co.)	Dothan, Ala.	West.	Bechtel	PWR	(11)	821	2,652
27-1	Kewaunee Nuclear Power (Wisconsin Power & Light Co., Wisconsin Public Service Co. and Madison Gas & Electric Co.)	Carlton, Wis.	West.	Pioneer	PWR	(4)	535	1,650
28-1	La Crosse (Genoa) Nuclear Generating Station (Dairyland Power Cooperative)	La Crosse, Wis.	AC	S&L	BWR	(1)	50	165
29-1	Maine Yankee Atomic Power Plant (Maine Yankee Atomic Power Co.)	Wiscasset, Maine	Comb.	S&W	PWR	(7)	790	2,500
30-1	Millstone Nuclear Power Station, Unit 1 (Northeast Nuclear Energy Co.)	Waterford, Conn.	GE	Ebasco	BWR	(2)	660	2,011
31-1	Millstone Nuclear Power Station, Unit 2 (Northeast Nuclear Energy Co.)	Waterford, Conn.	Comb.	Bechtel	PWR	(11)	830	2,560
32-1	Monticello Nuclear Generating Plant (Northern States Power Co.)	Monticello, Minn.	GE	Bechtel	BWR	(2)	545	1,670
33-1	Nine Mile Point Nuclear Station, Unit 1 (Niagara Mohawk Power Corp.)	Scriba, N.Y.	GE	S&W	BWR	(2)	610	1,850

CURRENTLY LICENSED REACTORS IN UNITED STATES (continued)

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufacturer **	Architect Engineer **	Reactor Type	Containment Type *	Power: Unit Size Net MW(e)	Power: Reactor MW (t)
34-1	North Anna Power Station, Unit 1 (Virginia Electric & Power Co.)	Mineral, Va.	West.	S&W	PWR	(7)	907	2,775
35-1	Oconee Nuclear Station, Unit 1 (Duke Power Co.)	Seneca, S. C.	B&W	Utility & Bechtel	PWR	(10)	887	2,568
35-1	Oconee Nuclear Station, Unit 2 (Duke Power Co.)	Seneca, S. C.	B&W	Utility & Bechtel	PWR	(10)	887	2,568
35-1	Oconee Nuclear Station, Unit 3 (Duke Power Co.)	Seneca, S. C.	B&W	Utility & Bechtel	PWR	(10)	887	2,568
36-1	Oyster Creek Nuclear Power Plant, Unit 1 (Jersey Central Power & Light Co.)	Toms River, N.J.	GE	B&R	BWR	(2)	650	1,930
37-1	Palisades Nuclear Plant, Unit 1 (Consumers Power Co. of Michigan)	South Haven, Mich.	Comb.	Bechtel	PWR	(10)	805	2,530
38-1	Peach Bottom Atomic Power Station, Unit 2 (Philadelphia Electric Co., Public Service Electric & Gas Co., Atlantic City Electric Co., and Delmarva Power & Light Co.)	Peach Bottom, Pa.	GE	Bechtel	BWR	(2)	1,065	3,293
38-1	Peach Bottom Atomic Power Station, Unit 3 (Philadelphia Electric Co., Public Service Electric & Gas Co., Atlantic City Electric Co., and Delmarva Power & Light Co.)	Peach Bottom, Pa.	GE	Bechtel	BWR	(2)	1,065	3,293
39-1	Pilgrim Nuclear Power Station, Unit 1 (Boston Edison Co.)	Plymouth, Mass.	GE	Bechtel	BWR	(7)	655	1,998
40-1	Point Beach Nuclear Plant, Unit 1 (Wisconsin Electric Power Co. and Wisconsin Michigan Power Co.)	Two Creeks, Wis.	West.	Bechtel	PWR	(10)	497	1,518
40-1	Point Beach Nuclear Plant, Unit 2 (Wisconsin Electric Power Co. and Wisconsin Michigan Power Co.)	Two Creeks, Wis	West.	Bechtel	PWR	(10)	497	1,518

CURRENTLY LICENSED REACTORS IN UNITED STATES (continued)

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufacturer **	Architect Engineer **	Reactor Type	Containment Type *	Power Unit Size Net MW(e)	Reactor MW (t)
41-1	Prairie Island Nuclear Generating Plant, Unit 1 (Northern States Power Co.)	Red Wing, Minn.	West.	Pioneer	PWR	(4)	530	1,650
41-1	Prairie Island Nuclear Generating Plant, Unit 2 (Northern States Power Co.)	Red Wing, Minn.	West.	Pioneer	PWR	(4)	530	1,650
42-1	Quad-Cities Station, Unit 1 (Commonwealth Edison Co. and Iowa-Illinois Gas & Electric Co.)	Cordova, Ill.	GE	S&L	BWR	(2)	789	2,511
42-1	Quad-Cities Station, Unit 2 (Commonwealth Edison Co. and Iowa -Illinois Gas & Electric Co.)	Cordova, Ill.	GE	S&L	BWR	(2)	789	2,511
43-1	Rancho Seco Nuclear Generating Station, Unit 1 (Sacramento Municipal Utility District)	Clay Station, Calif.	B&W	Bechtel	PWR	(11)	918	2,772
44-1	Robert Emmett Ginna Nuclear Power Plant, Unit 1 (Rochester Gas & Electric Co.)	Ontario, N.Y.	West.	Gilbert	PWR	(9)	490	1,520
45-1	Salem Nuclear Generating Station,Unit 1,2 (Public Service Electric & Gas Co., Philadelphia Electric Co., Atlantic City Electric Co., and Delmarva Power & Light Co.)	Salem, N.J.	West.	UE&C	PWR	(8)	1,090	3,338
46-1	San Onofre Nuclear Generating Station, Unit 1 (Southern California Edison and San Diego Gas & Electric Co.)	San Clemente, Calif.	West.	Bechtel	PWR	(3)	436	1,347
47-1	Shippingport Atomic Power Station (DOE and Duquesne Light Co.)	Shippingport, Pa.	West.	B&R,S&W	PWR	(3)	60	236
48-1	St. Lucie Plant, Unit 1 (Florida Power & Light Co.	Fort Pierce, Fla.	Comb.	Ebasco	PWR	(4)	802	2,560
49-1	Surry Power Station, Unit 1 (Virginia Electric & Power Co.)	Gravel Neck, Va.	West.	S&W	PWR	(7)	822	2,441

CURRENTLY LICENSED REACTORS IN UNITED STATES (continued)

Seismic Review Table No.	Name and/or owner	Location	NSSS Manufacturer **	Architect Engineer **	Reactor Type	Containment Type *	Power Unit Size Net MW(e)	Power Reactor MW (t)
49 -1	Surry Power Station, Unit 2 (Virginia Electric & Power Co.)	Gravel Neck, Va.	West.	S&W	PWR	(7)	822	2,441
50 -1	Three Mile Island Nuclear Station, Unit 1 (Metropolitan Edison Co.)	Middletown, Pa.	B&W	Gilbert	PWR	(10)	819	2,535
51 -1	Three Mile Island Nuclear Station, Unit 2 (Metropolitan Edison Co.)	Middletown, Pa.	B&W	B&R	PWR	(10)	906	2,772
52 -1	Trojan Nuclear Plant, Unit 1 (Portland General Electric Co., Eugene Water & Electric Board, and Pacific Power & Light Co.)	Prescott, Oreg.	West.	Bechtel	PWR	(12)	1,130	3,411
53-1	Turkey Point Plant, Unit 3 (Florida Power & Power Co.)	Florida City, Fla.	West.	Bechtel	PWR	(10)	693	2,200
53 -1	Turkey Point Plant, Unit 4 (Florida Power & Power Co.)	Florida City, Fla.	West.	Bechtel	PWR	(10)	693	2,200
54 -1	Vermont Yankee Nuclear Power Station (Vermont Yankee Nuclear Power Corp.)	Vernon, Vt.	GE	Ebasco	BWR	(2)	514	1,593
55 -1	Yankee-Rowe Nuclear Power Station (Yankee Atomic Electric Co.)	Rowe, Mass.	West.	S&W	PWR	(3)	175	600
56 -1	Zion Nuclear Plant, Unit 1 (Commonwealth Edison Co.)	Zion, Ill.	West.	S&L	PWR	(10)	1,040	3,250
56 -1	Zion Nuclear Plant, Unit 2 (Commonwealth Edison Co.)	Zion, Ill.	West.	S&L	PWR	(10)	1,040	3,250

* Containment types:
- (1) Pre-Mark (Steel)
- (2) Mark I (Steel)
- (3) Dry Containment-Spherical (Steel)
- (4) Dry Containment-Cylindrical (Steel)
- (5) Mark I (Reinforced Concrete)
- (6) Ice Condenser (Reinforced Concrete)
- (7) Sub-Atmospheric (Reinforced Concrete)
- (8) Atmospheric (Reinforced Concrete)
- (9) Without Buttresses (Pre-Stressed Concrete)
- (10) 6 Buttresses With Shallow Dome (Pre-Stressed Concrete)
- (11) 3 Buttresses With Shallow Dome (Pre-Stressed Concrete)
- (12) 3 Buttresses With Hemispherical Dome (Pre-Stressed Concrete)

** Manufacturers and Engineers
- AC = Allis-Chalmer Mfg. Co.
- AEP = American Electric Power Service Corp.
- B&R = Burns & Roe, Inc.
- B&W = Babcock & Wilcox Co.
- Comb. = Combustion Eng., Inc.
- GA = General Atomic
- GE = General Electric Co.
- G&H = Gibbs & Hills, Inc.
- S&W = Stone & Webster Eng. Corp.
- S&L = Sargent & Lundy Engineers
- TVA = Tennesse Valley Authority
- UE&C = United Engineers & Constructors
- West. = Westinghouse Electric Corp.

CURRENTLY LICENSED REACTORS IN UNITED STATES (continued)

SEISMIC REVIEW TABLE

Docket Number
50-313

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
ARKANSAS NUCLEAR UNIT No. 1 Reactor type: PWR Containment type: 3 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Babcock & Wilcox Arcitect Engineer: Bechtel	0.10	0.067	VII	0.20	0.133	A synthetic time history is generated so that its response spectra envelops the ground design spectrum.	Three components: 2 horizontal & 1 vertical. Each horizontal was combined with the vertical, assuming simultaneous occurrences.	SRSS (No closely spaced modes).	Housner	Time-history method Vertical ground response spectrum was used for equipment design (no vertical floor response spectra generated).
12-68/5-74	Sec. 5.1.1.2.5 p. 5-28a		p. 2-19	Sec. 5.1.1.2.5 p. 5-28a		p. 5.A-6 Amend. 28	Sec. 5.A.4.1 p.5.A-5 Amend. 28	Sec. 5.A.4.2 p. 5.A-7	Sec. 5.A 4.1 p. 5.A-5 Figs. 5.A-1 and 5.A-2	Sec. 5.A.4.2 p. 5.A-6 p. 5-28c Amend. 23

8/18/72

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Flat Slab 9 feet "All Class I structures utilize the shale bedrock as a foundation" Sec. 5.1.1.1 p. 5.1 Sec. 2.7.2 ~~p. 2-16~~	Bedrock which consists of Pennsylvanian McAlester formation shale. p. 2-24	13 ft to 24 ft. p. 2-16	Properties of shale, 10,000 to 14,500 fps. Table 2-5 p. 2-28	Most wells drilled into bedrock are less than 150 ft. Sec. 2.5.3 p. 2-7a	Not available.	Stick model with soil springs, as indicated in Fig. 5A-3 Fig. 5A-4 Fig. 5A-5	Not available	Unclear information Sec. 5.1.1.5.6 p. 2-28a	Not availabl

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE		DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
	(% critical damping)	Y = 1/ϕ (1.25 D + 1.0 R + 1.25 E) Y = 1/ϕ (1.25 D + 1.25 H + 1.25 E) Y = 1/ϕ (1.25 D + 1.25 H + 1.25 W) Y = 1/ϕ (1.0 D + 1.8 E) (For structural element carrying mainly earthquake forces.) Y = 1/ϕ (1.0 D + 1.0 R + 1.0 E') Y = 1/ϕ (1.0 D + 1.0 H + 1.0 E') (0.9 D is used where dead load subtracts for critical stress in the first three equations.) Y = yield strength. D = dead load. R = force or pressure on structure due to rupture of any pipe. H = force on structure due to thermal expansion. E = design earthquake load. E' = maximum earthquake load. W = tornado load ϕ = 0.9 for reinforced concrete, 0.85 for shear, bond. Anchorage in reinforced concrete. 0.75 for spirally reinforced concrete component members. 0.70 for tied component members. 0.90 for fabricated structural steel, and 0.90 for reinforced steel (not prestressed) in direction of tension.	ACI-318-63 Code AWS D12.1-61 Ultimate strength design "Design of Protective Structures", Dept. of Navy, NP-3726, August 1950.
Welded steel plate assemblies	1.0/1.0		
Welded steel framed structures	2.0/2.0		
Bolted or riveted steel framed structure	2.5/2.5		
Reinforced concrete equipment supports	2.0/3.0		
Reinforced concrete frames and buildings	3.0/5.0		
Prestressed concrete structure	2.0/5.0		
Sec. 5.A.4 p. 5.A-6		Sec. 5.A.3 p. 5.A-3 p. 5.A-4	Sec. 5.A.3 p. 5-38a p. 5.A-3 Amend. 28

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA		
		LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
(% critical damping) Steel piping 0.5/0.5	Analytical and/or testing	L. C. for Internals, vessels, integral support attachments and piping:		ASME BPVC, Section III ANSI B31.7 Nuclear Power piping code -
		<u>L.C.</u>	<u>Stress Limit</u>	
		Design loads + design earthquake loads	$P_M \leq 1.0\ S_M$ $P_L + P_B \leq 1.5\ S_M$	
		Design loads + SSE	$P_M \leq 1.2\ S_M$ $P_L + P_B \leq 1.2\ (1.5\ S_M)$	
		Design loads + pipe rupture	$P_M \leq 1.2\ S_M$ $P_L + P_B \leq 1.2\ (1.5\ S_M)$	
		Design loads + SSE	$P_M \leq 2/3\ S_u$ $P_L + P_B \leq 2/3\ S_u$	
Sec. 5A.4 p. 5.A-6	Sec. 5.A.4.2 p. 5A-6 p. 5A-8	Sec. 4.1.2 p. 4-4		Sec. A-.3 p. A-2

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Class I electrical equipment is seismic qualified in accordance with the IEEE Guide for seismic qualification of Class I electrical equipment for nuclear power generating stations, JcNPS/Sec. 5 (to be designated IEEE 344). Sec. 8.1 p. 8-1, Amendment No. 22, December 14, 1971

SEISMIC REVIEW TABLE

Docket Number
50-368

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Arkansas Nuclear One Unit No. 2 Reactor type: PWR Containment type: 3 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Bechtel	0.10	0.067	VII	0.20	0.133	Synthetic time history	Three components: two horizontal and one vertical. Each horizontal was combined with the vertical, assuming simultaneous occurrence.	SRSS	Design response spectra generated from time-histories as per AEC Reg. Guide 1.60 (BC-TOP-4)	Time-history method using synthetic earthquake acceleration time history
12-72/9-78	p. 2.5-25	p. 3.7-7		p. 2.5-25	p. 3.7-7	pg. 3.7-1	p. 3B-1	p. 3.7-9	p. 3.7-1	p. 3.7-3

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete flat circular slab. Depth not available.	Moderate to stiff, plastic, red and tan clay with occasional zone of silty clay, which *overlies black, dense*, horizontally bedded shale and interbedded shale and sandstone of the McAlester formation.	70 ft to 90 ft.	Not available.	About 10 ft below ground surface.	Ozark Dam Dardanelle Dam Robert S. Kerr Dam	Stick model **with fixed base**	Not available.	No soil damping	Not available.
p. 3.8-46	p. 2.5-9	p. 2.5-8		p. 2.5-11	p. 2.4-6 to 2.4-8	p. 3.7-3		p. 3.7-2	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE		DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
	(% critical damping)	A. Design loading case: 1) $D+L+F+T_o$ 2) $D+L+F+P+T_A$ B. Factored loading case: 1. $C = 1/\phi\ ((1.0+0.05)\ D + 1.5\ P + 1.0\ T_A + 1.0\ F)$ 2. $C = 1/\phi\ ((1.0+0.05)\ D + 1.25\ P + 1.0\ T_A + 1.25\ H + 1.25\ E + 1.0\ F)$ 3. $C = 1/\phi\ ((1.0+0.05)\ D + 1.25\ H + 1.0\ R + 1.0\ F + 1.25\ E + 1.0\ T_o)$ 4. $C = 1/\phi\ ((1.0+0.05)\ D + 1.0\ F + 1.25\ H + 1.0\ W' + 1.0\ T_o)$ 5. $C = 1/\phi\ ((1.0+0.05)\ D + 1.0\ P + 1.0\ T_A + 1.0\ H + 1.0\ E' + 1.0\ F)$ 6. $C = 1/\phi\ ((1.0+0.05)\ D + 1.0\ H + 1.0\ R + 1.0\ E' + 1.0\ F + 1.0\ T_o)$ C = Required capacity of the containment D = Dead loads. E = Operating basis earthquake loads. E' = Design basis earthquake loads. F = Prestress loads. H = Pipe expansion loads. L = Live loads. P = LOCA pressure loads. R = Pipe rupture loads T = LOCA thermal loads. T_o = Operating thermal loads. W' = Tornado wind and tornado missile loads. ϕ = Capacity reduction factors.	ACI 318-63 AISC 1969 Supplement 1, 2, November 1970 and December 1971.
Welded steel frame structures	2.0/5.0		
Bolted and riveted steel	3.0/5.0		
Reinforced concrete structure and equipment supports	3.0/5.0		
Prestressed concrete structures	2.0/5.0		
Bolted or riveted steel frame structures	2.5/2.5		
p. 3.7-15		p. 3.8-7 to 3.8-8	p. 3.8-3

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
(% critical damping) Steel piping 0.5/0.5 Vital piping 0.5/1.0 Welded steel plate assemblies 1.0/1.0	Analytical	Loading combination 1: normal operating loads + OBE loads. Loading combination 2: normal operating loads + DBE loads. Loading combination 3: normal operating loads + DBE loads + pipe rupture loads.	ASME BPVC Section III
p. 3.7-15	p. 3.6-6	p. 3.6-4	p. 3.6-4

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Equipment supplied by NSSS vendor: Combustion Engineering Topical Report CENPD-61 Equipment supplied by other than NSSS vendor: IEEE Standard 344-1971 p. 3-10.2

SEISMIC REVIEW TABLE

Docket Number
50-334

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Beaver Valley Power Station Unit No. 1 Reactor type: PWR Containment type: Sub-atmospheric (Reinforced concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Stone & Webster	0.06	0.04	IV	0.125	0.085	Compared with El Centro 1940 and Taft 1952, Golden Gate 1957.	Three components. Combination is simultaneous.	SRSS	Housner response spectra was generated which enveloped El Centro, Taft and Golden Gate time histories. Performed by Dr. R. V. Whitman	Time-history method.
6-70/7-76	Sec.2.5.3 p. 2.5-4	Sec.2.5.3 p. 2.5-4	Sec.2.5.3 p. 2.5-3	Sec.2.5.3 p. 2.5-4	Sec.2.5.3 p. 2.5-4	Sec. 2.6.4.2 p. 2.6-11	Q. 3.15-5 Amend. 5 10/10/73 App. B.1-3	Q. 3.15-1 Amend. 1 4/23/73	Figs. 2.5-1 and 2.5-2 Pg. 2.5-3 App. 2D	

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat 10 ft thick	Gravel terrace	100 ft	Varying from 800 to 1250 psf	10 ft to 50 ft average 30 ft below surface.	3.1 miles downstream from Montgomery Lock and Dam 19.6 miles upstream from New Cumberland Rock and Dam.	Stick model with soil springs.	(1) Containment structure G = 22,000 psi (2) Fuel building, auxiliary building and other near surface building G = 17,000 psi (3) Intake structure G = 17,000 psi	Not available.	5% OBE 7% DBE
Sec. 2.6.3.1 p. 2.6-3	Sec. 2.4 p. 2.4-2	Sec. 2.4 p. 2.4-2	Sec. 2.6.2.3 p. 2.6-3	Sec. 2.3.2.1.1 p. 2.3-3	Sec. 2.3.1 p. 2.3-1	Sec. 2.6.4.4 p. 2.6-15	Sec. 2.5.3 p. 2.5-5		App. B pg. B.1-3

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment structure	5.0/7.0	Concrete structure D.L. + L.L. D.L. + L.L. + OBE D.L. + L.L. DBE D.L. + L.L. + TOR D.L. + L.L. + F	Using working stress design ACI 318-63
Steel reinforced concrete (no cracking)	0.5 to 1.0		
Welded steel, well reinforced concrete (with slight cracking)	2.0	Steel structure D.L. + L.L. D.L. + L.L. + OBE D.L. + L.L. + DBE D.L. + L.L. + TOR D.L. + L.L. + F	Steel structure, AISC-63, Part I Specified minimum yield strength for structural steel.
Reinforced concrete (with considerable cracking)	2.0		
Bolt steel	5.0		
Welded steel	5.0		
Reinforced concrete	5.0		
Bolted steel	7.0		
Amendment I, Sec. B.1.2, Table B.1-3, p. B.1-3 4/23/73		Amendment VII, p. B.1-6 (3/29/74)	Amendment VII, P. B.1-7 3/29/74

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping	0.5/1.0	Analytical and testing.	Pressure piping 1. Normal conditions (a) $P_m \leq S$ (b) P_m (or P_L) $\leq S$ 2. Upset conditions (a) $P_m \leq 1.2\ S$ (b) $P_m + P_B \leq 1.5\ S$ 3. Emergency conditions (a) $P_m \leq 1.2\ S$ (b) $P_m + P_B \leq 1.5\ (1.2\ S)$ Pressure vessel 1. Normal conditions (a) $P_m \leq S_m$ (b) $P_m + P_B \leq 1.5\ S_m$ (c) $P_m + P_B + Q \leq 3\ S_m$ 2. Upset conditions (a) $P_m \leq S_m$ (b) $P_m + P_B \leq 1.5\ S_m$ (c) $P_m + P_B + Q \leq 3\ S_m$ 3. Emergency conditions (a) $P_m \leq 1.2\ S_m$ (b) $P_m + P_B \leq 1.5\ (1.2\ S_m)$	Piping ANSI, B31.1 pressure piping code with diameters of 6 in. NPS and below. ASME BPVC, Section III (1968 edition)
Amendment I, Table B.1-3 4/23/73			For further details refer to Table B.3-4	Question 3.22-1

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Testing for mounted components	"Class I instrumentation and electrical equipment are designed to maintain their capability to: 1. Initiate a protective action during DBE and OBE 2. Withstand seismic disturbances during post accident operation	IEEE STD 344-1971 "Seismic Qualification of Class I Electric Equipment for NPP Generating Station". p. B. 2-14

SEISMIC REVIEW TABLE*

Docket Number
50-155

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Big Rock Point Nuclear Plant Reactor type: BWR Containment type: Pre-Mark (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel 5-60/8-62	Not used	Not used	Not available	.05 and 0.025 (see last column of this page) 0.12 for RDS only. Sec. 2-11	not used	not used	one horizontal component 3 directions with SRSS for reactor depressurization system only Sec. 2-11	SRSS for RDS only	Not used	The lateral concrete loads for design of internal concrete structures were determined from U.B.C. requirements. A seismic factor of 0.025 was used for the equivalent lateral coefficient for these structures as well as other major structures, e.g. turbine building, 240 ft. high stack, control room and waste storage building. RDS reanalyzed in 1974 using R.G. 1.60, floor response spectra by Kapur method.

*Information obtained from BNL Docket search and SEPB Report prepared by LLL; EDAC Report #175-130.04, January 1979.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
The lower segment of the spherical steel vessel is embedded in concrete and the structure extends 27 ft. below grade. The foundation consists of a combination of a 3-foot thick concrete mat and reinforced concrete footings from 38 ft. to 8 ft. below grade.	Rock	Not available	Not available	Not available	Not available	Not used	Not available	Not used	Not avail-able

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% Critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment: 4.0 used in 1974 reanalysis of reactor depressurization system to acceleration equal to 0.12g. RDS components assumed to have damping values of R.G. 1.61.	Containment: Seismic (0.05g) + DL + snow Internal Concrete Structure: Seismic (0.05g) + DL + equipment NSSS: Seismic (0.05g) + DL + pressure NSSS Piping: Seismic (0.025g) + pressure + equipments Turbine Building: Seismic (0.025g) + DL + equipment Sec. 3-3	Containment: ASME B and PV Sec. VI, VIII, IX UBC - 1958 ACI - 318-56 Sec. 2-11

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
		DESIGN CRITERIA	
DAMPING OBE/SSE (% Critical damping)	METHOD OF QUALIFICATION	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Containment/Reactor Vessel: ASME BPVC Sec. II, VI, VIII, IX, 1958 Piping and Supports: ASA B 31.1 1955

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE (% Critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
	Test		MIL-STD-167, Mechanical vibration of shipbord equipment MIL-STD-901C, Requirements for shock test. "Seismic qualification of RDS for BRP plant". Amend. 8, Docket 50155-50

SEISMIC REVIEW TABLE

Docket Number
50-259, 260, 348

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Browns Ferry Nuclear Plant Unit Nos. 1, 2, & 3 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Tennessee Valley Authority	0.10	0.067	VII	0.20	0.13	Design spectra compared with the El Centro, May 1940, N-S component, normalized to maximum acceleration. El Centro time history enveloped ground spectrum and was used in time-history analyses	Three components: Each horizontal component combined with vertical component simultaneously. "A vertical acceleration is considered to act simultaneously (with horizontal) and to increase or decrease the vertical load, whichever is most conservative.	SRSS	Housner design spectra	Time-history method.
Unit 1: 5-67/6-73 Unit 2: 5-67/6-74 Unit 3: 7-68/8-76	Sec. 2.5.4 p. 2.5-6	p.12.2-2	p. 2.5-6	Sec. 2.5.4 p. 2.5-6	p.12.2-2	Sec. 2.5-4 pp. 2.5-7, 2.5-8, 2.5-12	p. 12.2-32 Sec. C.3-2 p. C.0-3	Sec. C.3-2 p. C.0-3	Figs. 2.5-15 and 2.5-16 , 2.5-17 p. 2.5-7	Sec. 12.2.2.8 p. 12.2-12

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
[B]ase slab with a [c]ircular mass of [c]oncrete at the [c]enter supporting [t]he drywell.	Bedrock	Average depth 54 ft (41 to 69 ft)	Not available.	Ground water is derived from precipitation.	Wheeler Dam Wilson Dam	Lumped mass model with soil springs	2,300,000 psi bedrock	Not available	5% for all modes
	Tuscombia formation	50 ft below bedrock							
	Fort Payne formation	145 ft below Tuscombia							
Sec. 12.2.2.1 p. 12.2-1	Sec. 2.5.2.3.2 pp. 2.5-1&2.5-2			Sec. 2.4.2.1 p. 2.4.1	p. 2.4-3	Sec. 12.2.2.8 p. 12.2-11 p. 12.2-69 Fig. 12.2-78	Sec. 2.5.2.4.2 p. 2.5-5	p. 12.2-69	Sec.12.2.28 p. 12.2-31

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>Steel structure 1.0

Concrete 5.0

Sec. 12.2.2
p. 12.2-4</td><td>These loads are considered in the following combinations:
Reactor building
Case 1. Prestartup - DL+LL+P
Case 2. Operating - DL+LL+P+THERM+RESTR
Case 3. Operating + Earthquake
-A. DL+LL+P+THERM+RESTR+OBE
-B, DL+LL+P+THERM+RESTR+DBE

where

DL = dead load
LL = live load
P = pressure transmitted through polyurethane foam at operating temperature
OBE = Operating Basis Earthquake (0.1 g)
DBE = Design Basis Earthquake (0.2 g)
THERM = thermal load at operating temperatures
RESTR = restraint to thermal growth of shield by pools

For more details: refer to Tables 12.2-1 through 12.2-43

Sec. 12.2.2.2.3
p. 12.2-4</td><td>ACI-318-63
N.O. + OBE $\leq 0.5\ f_y$
N.O. + DBE $\leq 0.85\ f'_c$ or $0.9\ f_y$

Ultimate strength method

AEC Q. 12.2-10
p. 12.2-4</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
			LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping	0.5	Analytical	Deformation limit Primary stress limit Buckling stability limit Fatigue limit For details refer to Tables C.0-1 to C.0-7.	Table C.0-1 Table C.0-2 Table C.0-3 Table C.0-4	Piping ANSI B31.1.0 ANSI B31.7 Vessel ASME BPVC, Section III
Equipment	1.0				
Sec. C.3-2 p. C.0-3		Appendix C Section C.3	Section C.2-6 p. C.0-2		Appendix C Section C.4-1

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-324, 325

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA: OBE HOR. g	OBE VERT. g	SSE INTENSITY MM	SSE HOR. g	SSE VERT. g	EARTHQUAKE TIME HISTORY	METHOD OF COMBINATION: NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	DESIGN SPECTRA: TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
Brunswick Steam Electric Plant Units 1 & 2 Reactor type: BWR Containment type: Mark I (Reinforced concrete) NSSS Manufacturer: General Electric Architect Engineer: United Engineers & Constructors	0.08	0.053	VII (SSE)	0.16	0.107	1940 N-S El Centro spectrum normalized by a factor was used for developing the design spectra.	Three components, each horizontal was combined with the vertical, resulting in two distinct load cases.	For piping equipment by SRSS C.4.3.2 For structure absolute sum.	The envelope of the Housner spectra and the El Centro spectra was termed as the smoothed 1940 N-S El Centro normalized spectrum. Fig. 2.6-7 Fig. 2.6-9	Time-history method
Unit 1: 2-70/10-76 Unit 2: 2-70/12-74	Sec. 2.6 p. 2.6-6	Sec. 2.6 p. 2.6-10	Sec. 2.6 p. 2.6-11	Sec. 2.6 p. 2.6-7	Sec. 2.6 p. 2.6-11	Sec. 2.6.6.1 p. 2.6-10	C4.3.2 p. C-56	Comment C-10 MC.10-1 Amend. 14 1972	Sec. 2.6 p. 2.6-9 Fig. 2.6-7	Comment C.3, P.MC.3-1 Amend. 13 (Sept. 72)

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat foundation, founded on a strata of very dense-fine to medium-coarse sand. Depth not available.	Sand and clay. Limestone Hard calcareous clay and cretaceous rock. Crystalline	115 ft 115 ft down to 1500 ft	Thick. (ft) / V_s (ft/sec) 35 / 750 30 / 1400 43 / 5500 127 / 4500 1290 / 3000	Table M.2.17-1 gives ground water details.	Not available.	Lumped mass with soil springs. See design reports 4, 9, and 10.	Not available	Soil structure interaction damping .04/.07 critical damping for OBE/DBE.	Not available.
Sec. 12.2.1 p. 12.2-1	Sec. 1.5 p. 1.5-2	Sec. 1.5 p. 1.5-2	Fig. 2.6-7	Comment 2.17 PM2.17-1 Amend. 14, 11/72		C.57, p. MC.57-1		Table C-1	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	**DESIGN CRITERIA** **LOAD COMBINATION**	**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Reinforced concrete: Primary containment Other Class I structures Steel structures: (Reactor building and other Class I structures) Bolted or riveted Welded Table C-1	 4.0/7.0 4.0/7.0 5.0/10.0 2.0/5.0	Primary containment (Drywell & Suppression Chambers) $U_1 = (1.0 \pm 0.1)\ D + 1.50\ P + 1.0\ T_{1.5} + 1.0\ R$ $U_2 = (1.0 \pm 0.1)\ D + 1.25\ P + 1.0\ T_{1.25} + 1.25\ E + 1.0\ R$ $U_3 = (1.0 \pm 0.1)\ D + 1.00\ P + 1.0\ T_{1.00} + 1.00\ E' + 1.0\ R$ $T_p = (1.0 \pm 0.1)\ D + 1.15\ P$ (Pressure test condition) Class I Structures $U = 1.5\ D + 1.8\ L + 1.0\ T + R + Pr$ $U = 1.5\ D + 1.5\ L + 1.5\ E + 1.0\ T + R + Pr$ $U = 0.9\ D + 1.5\ W + 1.0\ T + R + Pr$ $U = (1.0 \pm 0.1)\ D + 1.0\ E' + 1.0\ T + R + Pr$ $U = (1.0 \mp 0.1)\ D + 1.0\ W' + 1.0\ T + R + Pr$ $U = 1.5\ \overline{D} + 1.5\ L + 1.5\ W + 1.0\ T + R + Pr$ Sec. C.2.6.1 p. C-9	Codes ACI 318-63, Part IV B Ultimate strength design AISC (1963) specification for the erection of structural steel Plant stack design, ACI 307-69 Comment 22 p. MC.22-1 Amendment 13 (Sept. 1972) C-5

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA			
			LOAD COMBINATION			ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
			Piping Design condition	Load combination	Stress limits	
Equipment	1.0/2.0	Analytical and testing	Design, normal and upset	Pressure	S_h	ANSI B31.1 - 1967 Power piping ASME BPVC, Sec. III
Piping	0.5/2.0			Pressure; dead weight	S_h	Valves ANSI-B31.1-67 ANSI-B16.5
				Pressure, dead weight, OBE	1.25 S_h	
				Pressure, dead weight, thermal	S_n+S_h	Pumps ANSI-B31.1-67 ASME Sec. III. Class C
			Emergency	Pressure, dead weight, DBE	1.8 S_h	
Table C-1		Sec. 2.2 C-4	Table C-7 through C-29 Amendment 13, Comment 4.3, p. M4.3-1			Amendment 13 (Sept. 1972) p. M4.1-1 Sec. A.1.1, p. 2

p. MC.18-3

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and Testing	OBE Combined stresses $\leq 0.6\ S_y$. DBE Combined stresses $\leq 0.9\ S_y$.	IEEE 344-1971 Equip — Max. Hor. "g" Voltage pre-amp — 8.5 Temp. control switch — 12 Intermediate range monitor — 1.5 see Table C-30
	Sec. 2.2 p. C-4	Comment 7.8, p. M7.8-5 Amendment 13 (Sept. 1972)	Table C-30 Comment 7.8 p. M7.8-2 Amendment 13 (Sept. 1972)

SEISMIC REVIEW TABLE

Docket Number
50-317, 318

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Calvert Cliffs Nuclear Power Plant Units No. 1 & 2 Reactor type: PWR Containment type: 6 Buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Bechtel	0.08	0.053	VII	0.15	0.10	Compared with digitalized El Centro earthquake 1940 (E-W) normalized to: 0.08 g horizontal 0.053 g vertical	Horizontal and vertical components combined simultaneously.	SRSS including closely spaced modes.	1. Housner spectra for frequency >0.33 cps. 2. Newmark spectra for frequency <0.33 cps (Figs. 2.6-4, and 2.6-5)	"Digitized El Centro was used in the analysis of Class I equipment. Class 2 structures use UBC Zone 3. AEC TID 7024 "Nuclear Reactors and Earthquakes".
Unit 1:7-69/7-74 Unit 2:7-69/11-76	Sec. 2.6.5.2 p. 2.6-9	Sec. 2.6.5.2 p. 2.6-9	← Sec. 2.6.5.3 p. 2.6-9 →			Sec. 2.6.5.4 p. 2.6-10	Sec. 5A.3.1.4 p. 5A-5	Sec. 5.1.3.2(b) p. 5-22	Sec. 2.6.5.4 p. 2.6-10	p. 2.6-10 p. 5A-6

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Foundation for containment: 10 ft thick reinforced concrete slab. Sec. 5.1.2.1 p. 5.2	Major structure: Miocene sandy and clay silts of Chesapeake group. Appurtenant structure: surficial pleistocene silt which overlies the miocene sediments. Sec. 2.6.5.1 p. 2.6-9	200 ft Sec. 2.4.1 p. 2.4-1	1600 fps Sec. 2.6.4.4 p. 2.6-7	Varies from 8 ft to 82 ft. Sec. 2.5.3.3 p. 2.5-9	Not available.	Stick model with soil springs. Sec. 5.1.3.2 p. 5-21	Not available.	Soil: % critical damping OBE: 2% SSE: 3% Rocking Motion Prestressed concrete $\frac{E'}{7\%}$ Rocking Motion Reinforce concrete 7% Sec. 5A.3.1.4 p. 5A-5, p. 5A-6	Not available

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
	(translational)	$Y \geq 1/\Phi\ (1.05\ D + 1.5\ P + 1.0\ T_A + 1.0\ F)$	
1. Welded steel framed structure	1.0/1.0	$Y \geq 1/\Phi\ (1.05\ D + 1.25\ P + 1.0\ T_A + 1.25\ H + 1.25\ E + 1.0\ F)$	ACI-318-63, when Φ is taken as 1.
2. Bolted or riveted steel framed structure	2.5/2.5	$Y \geq 1/\Phi\ (1.05\ D + 1.25\ H + 1.0\ R + 1.0\ F + 1.25\ E + 1.0\ T_o)$ $Y \geq 1/\Phi\ (1.05\ D + 1.25\ H + 1.0\ F + 1.25\ W + 1.0\ T_o)$	
3. Reinforced concrete frames and buildings	3.0/5.0	$Y \geq 1/\Phi\ (1.0\ D + 1.0\ P + 1.0\ T_A + 1.0\ H + 1.0\ E' + 1.0\ F)$	
4. Prestressed concrete structures	2.0/5.0	$Y \geq 1/\Phi\ (1.0\ D + 1.0\ H + 1.0\ R + 1.0\ E' + 1.0\ F + 1.0\ T_o)$	
	(rotational)	Y = Yield strength. D = Dead load. E = OBE E' = SSE	
1. Rocking motion for prestressed concrete structures	5.0/7.0	W = Tornado wind load. P = LOCI pressure load. F = Final prestress load.	
2. Rocking motion for reinforced concrete structures	5.0/7.0	T_A = Thermal load incident temperature gradient through walls and expansion liner R = Force or pressure on structure due to rupture of one pipe. H = Thermal expansion force. T_o = Thermal load due to normal operating temperature gradient through walls. Φ = Reduction factor.	
Sec. 5A.3.1.4 pp. 5A-5 and 5A-6		Sec. 5A.3.1.2 pp. 5A-3 and 5A-4	Sec. 5A.3.1.2 p. 5A-3

MECHANICAL & PIPING			
DAMPING OBE/SSE	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA** **LOAD COMBINATION**	**DESIGN CRITERIA** **ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
(% critical damping) (Translational) Steel piping 0.5/0.5 Welded steel plate assemblies 1.0/1.0	Analytical	Vessels / Piping 1. Design loading + OBE: Vessels: $P_m \le S_m$, $P_B+P_L \le 1.5\ S_m$; Piping: $P_m \le S_m$, $P_B+P_L \le 1.5\ S_m$ 2. Normal operating + SSE: Vessels: $P_m \le S_D$, $P_B \le 1.5\left[1-\left(\frac{P_m}{S_D}\right)^2\right]S_D$; Piping: $P_m \le S_D$, $P_B \le \frac{4}{\pi}S_D\cos\left(\frac{\pi}{2}\cdot\frac{P_m}{S_D}\right)$ 3. Normal operating + SSE + pipe rupture: Vessels: $P_m \le S_L$, $P_B \le 1.5\left[1-\left(\frac{P_m}{S_L}\right)^2\right]S_L$; Piping: $P_m \le S_L$, $P_B \le \frac{4}{\pi}S_L\cos\left(\frac{\pi}{2}\cdot\frac{P_m}{S_D}\right)$ P_m = Calculated primary membrane stress. P_B = Calculated primary bending stress. P_L = Calculated primary local membrane stress. S_m = Allowable stress limit ASME BPVC III. S_Y = Yield at temperature ASME BPVC III. S_D = Design stress. S_L = $S_Y + 1/3\ (S_U-S_Y)$ S_U = Tensile strength at temperature.	Reactor vessel: ASME BPVC III Piping: ASME BPVC III (1967) USAS B 31.7, Class I (Code cases 83, 1477 are included).
Sec. 5A.3.1.4 p. 5A-5	p. 5A-5	Sec. 4.2.1, Table 4-2 pp. 4-5 to 4-7	Sec. 4.2.1, Table 4-2 p. 4-7

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	"All electrical-systems and components vital to plant safety, including the emergency diesel generators, are designed as Class I so their integrity is not impaired by the design basis earthquake, high winds, or disturbances on the external electrical system". pg. 8.1	Not available

SEISMIC REVIEW TABLE

Docket Number
50-298

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Cooper Nuclear Station Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Burns & Roe, Inc.	0.10	0.05	VII	0.20	0.10	The accelerogram of the N69W component of the July 21, 1952 Kern County earthquake recorded at Taft, California was used to develop response spectra	Three components: two horizontal and one vertical. Each horizontal was combined with the vertical simultaneously.	Reactor vessel internals: SRSS for response spectrum method; algebraic sum for time-history method	Design spectrum response curves generated from 1952 Taft earthquake	Time-history method.
6-68/1-74	Vol. 1 Sec. 5.2.3 p. II-5-4	Vol. 1 Sec. 5.2.3 p. II-5-4	Vol. 1 Sec. 5.2.1 p. II 5-3	Vol. 1 Sec. 5.2.3 p. II-5-4	Vol. 1 Sec. 5.2.3 p. II-5-4	Vol. 1 Sec. 5.2.4 p. II-5-4	App. C Sec. 3.3.3 p. C-3-12	Vol. 1 Sec. 3.5.3 p. III-3-12	Vol. 1, Sec. 5.2.4 p. II-5-4, Figs. II-5-7 to II-5-10	Vol. VII Amend 9 Q.12.35

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Mat foundation. Depth not available.	Dense structure fill extending from the bed-rock surface to the mat foundation. Silty sand, sand silt, silt clay, clay.	Dense structure: not available. Silty sand: 10 to 25 ft.	Not available.	Not available.	Not available.	Stick model with rocking springs. No vertical or horizontal soil springs were included	Not available	5%	Not available.
Vol. I Sec. 5.2.3 p. II-5-4	Vol. I Sec.5.1.4, p. II-5.3	Vol. I Sec. 5.1.4,p.II-3				Vol. VII Amend 13 Q.12.55		Vol. V Appendix C p. C-2-7	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete structures. Steel frame structures. Welded assemblies. Bolted and riveted assemblies	5.0/7.0 2.0 1.0 2.0	D+E D+R D+R+E D+E+Flood D+T D+R+E' D = Dead load of structure and equipment. R = Loads resulting from jet forces and pressure and temperature due to rupture of a single pipe. E = OBE E' = SSE Flood = Loads due to flooding. W = Wind loads. T = Tornado loads.	ACI-318-63 for reinforced concrete. AISC Manual of Steel Construction (Sixth Edition)
Vol. IV, p. XII-2-16 Table XII-2-5		Appendix C Sec. 2.2 p. C-2-1	Vol. V Sec. 2.4 p. C-2-3

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital piping system	0.5	Analytical and Testing	Deformation limit Table C-3-2 Primary stress limit Table C-3-3 Buckling stability limit Table C-3-4 Fatigue limit Table C-3-5 Loading criteria Table C-3-7	Reactor vessel ASME BPVC, Sec. III Vol. V, Table C-3-7, p. C-3-14 Piping USAS, B31.1.0
Vol. IV, p. XII-2-16 Table XII-2-5		Vol. V, Sec. 3.3.26 & 3.3.2, p. C-3-11 & C-3-12. Appendix C Vol. VII p. 12.61.1	p. C-3-3, p. C-3-14, Table C-3-7 App. C, Table C-3-7	Vol. V, Table C-3-7, p. C-3-28

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number

50-302

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Crystal River Nuclear Generating Plant, Unit 3 Reactor type: PWR Containment type: Mark I (steel) NSSS Manufacturer: Babcock & Wilcox Architect Engineer: Gilbert Associates 9-68/12-76	0.05 Sec. 5.2.1.2.9 p. 5-12	0.033	V Sec. 2.5.4.1 p.2-31 Amend. 34 (11-15-73)	0.10 Sec. 5.2.1.2.9 p. 5-12	0.067	Response spectrum method was used in design. Floor response spectra generated with either 1940 N-S El Centro, 1952 N21E Taft or other time-history. Ref. Sec. 5.4.5 p.5-65A Amend. 26 (5-25-73) and Sec. 2.5.4.1 p. 2-31	Three components: Each horizontal combined with the vertical by absolute sum, although "structural response due to vertical input was assumed to be insignificant". Sec. 5.4.5 p.5-65 Amend. 26 (5-25-73)	SRSS Sec. 5.4.5.2 p.5-66 D Amend. 32 (10-1-73)	Spectra developed were estimated by two methods: Housner and Estere and Rosenblueth p. 2-32	Approximate method not based on time-history GAI Topical 1729 Sec. 5.4.5 p. 5-65A Amend. 26 (5-25-73)

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
For reactor building Mat foundation thickness 12.5 ft.	Natural soil: Laminated organic sandy silts and clays, interspersed with a pleistocene marine deposits. Underlying limerock Sec 2, p.2.1	Average of thickness of approximately 4 ft.	Not available	Depth of approximately 10 ft. below ground surface. Based on a ground datum of 100 ft. groundwater levels were recorded to rise approximately 1.5 ft. at peaks of high tides.	Not available.	Stick model with fixed base. Soil spring model was used to check accuracy of fixed base model.	Not available.	"Sum of material and radiation damping was assumed as small as 5%."	Not available.
	Bedrock: biogenic carbonates of tertiary age.	Approximately 20 ft. beneath the present ground surface.							
Sec. 2.5.7 p. 2-36 and Sec. 5.2, p.5-7 Amend.26,(5-25-73)				Sec. 2.5 p. 2-20 and Sec. 2.5.3.5 p. 2-29 and p.2-30		Sec. 5.4.5.2 p. 5-66 and Sec. 5.4.5 p. 5-65 and p. 5-65a Amend. 32 (10-1-73)		p. 5-65a	

Sec. 2.5.3
p 2-22

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	**DESIGN CRITERIA** — LOAD COMBINATION	**DESIGN CRITERIA** — ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reactor building shell Concrete support: Structure (Inside reactor building) Steel assemblies and structure a) bolted b) welded Other concrete structure (Above ground) p. 5-42	2.0 2.0 2.5 1.0 5.0	a) c= (1.0 ± .05) D + 1.5P + 1.0T b) c= (1.0 ± .05) D + 1.25P + 1.0T′ + 1.25 (E or W) c) c= (1.0 ± .05) D + 1.0P + 1.0 $\underline{T}$ + 1.0E′ d) c= (1.0 ± .05) D + 1.0 W_T + 1.0 P_t D= Dead load P= Design accident pressure load E= Seismic load based on 0.05g. E′= Seismic load based on 0.10g. W_T= Wind load based on Tornado P_t= Pressure load based on external pressure drop of 3 psig between inside and outside of reactor building. Sec. 5.2.3.2.1 p. 5-32	Reactor building: R. C. ACI 318-63 Structure concrete ACI 301-66 Structure steel AISC. 1963. Sec. 5.2.3.1 p. 5-31

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA** **LOAD COMBINATION**	**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Vital piping systems. 0.5 , p. 5-42	Analyses and test. Details. Ref. Table 5-5 p.5-86 AMEND. 17 (4-10-72) Sec. 5.4.5 p. 5-65 AMEND. 40 (7-3-74) p.5-64b AMEND. 45,(7-14-75)	For piping: primary stress + OBE $\leq 1.2 \times S_h$ thermal stress $\leq S_A$ where S_A = t $(1.25 S_c + 0.25 S_h)$ S_A = allowable stress S_h = basic material allowable stress at max. (hot) temp. S_c = basic material allowable stress at min. (cold) temp. p. 5-641 Amend. 45(7-11-75) and p. 5-63 Case / Load Combination / Stress Limits I) Design loads + design earthquake loads — $P_m \leq 1.00\ S_m$; $P_L + P_b \leq 1.5\ S_m$ II) Design loads + maximum hypothetical earthquake loads — $P_m \leq 1.2\ S_m$; $P_L + P_b \leq 1.2\ (1.5\ S_m)$ III) Design loads + pipe rupture loads — $P_m \leq 1.2\ S_m$; $P_L + P_b \leq 1.2\ (1.5\ S_m)$ IV) Design loads + maximum hypothetical earthquake loads — $P_m \leq 2/3\ S_y$; $P_L + P_b \leq 2/3\ S_y$ P_L = Primary local membrane stress intensity P_m = Primary general membrane stress intensity P_b = Primary bending stress intensity ~~S_m = Allowable membrane stress intensity~~	Reactor coolant system: ASME, boiler and pressurizer Vessel code, Sec. III, Art. 9 Summer, 1967 For piping (belongs to reactor coolant) USAS Sec. B31.7 Amendment 48,(3-16-76) p. 5-64a

S_u = Ultimate stress for unirradiated material at operating temperature

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Test or protype test or calculation. Ref. Sec. 7.1.3.1.4 p. 7-9b Amend. 32 (10-1-73)	Not available.	IEEE Standard. 314-1971 Sec. 7.1.1.8 p. 7-2b and p. 7-26 Amend. 45 (7-11-75)

SEISMIC REVIEW TABLE

Docket Number
50-346

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Davis-Besse Nuclear Power Station, Unit 1 Reactor type: PWR Containment type: Dry containment -cylindrical (steel) NSSS Manufacturer: Babcock & Wilcox Architect Engineer: Bechtel	0.08	.053	VII	0.15	0.10	E-W component of Helena Earthquake of October 31, 1935 was used as the basis for developing accelerograms of the OBE & DBE.	3 components: each horizontal combined with the vertical resulting two seismic load cases.	SRSS	Design spectrum response curves were developed by Newmark's method modifying the spectral amplification factors.	Time-history method.
	Vol. 1, Append. 2C Sec. D p. 2C-36	Vol. 1 Append. 2C Sec. D p. 2C-36	Vol. 1, Append. 2C, p. 2C-31	Vol. 1, Append. 2C, p. 2C-31	Vol. 1, Append. 2C, p. 2C-39	Vol. 1, Append. 2C, p. 2C-39	Vol. 2C, Sec. 3.7.1.6 p. 3-51	Vol. 2 Sec. 3.7.3.3 p. 3-63 Fig. 3-24	Vol. 1, Append. 2C, p. 2C-41 to 45	Vol. 2 Sec. 3.7.2 p. 3-54

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION: TYPE	BEARING INFORMATION: THICKNESS	BEARING INFORMATION: V_s PROFILE	GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
Main structure: Mat footings & Auxiliary building: Pier footings bearing on bed- [redacted] Depth not available.	Soil: Glaciolaguatrine and a till deposit.		For bedrock 5,700 fps to 7,500 fps	Prior to construction 571 ft. to 572 ft. (I.G.L.D.)	Not available.	Stick model with fixed base for the containment and the auxiliary building	Soil: For OBE: 10 $KIPS/ft^2$ For SSE: 12 $KIPS/ft^2$	Soil: For OBE 0.04 For SSE: 0.05	Not available.
	Bedrock: Tymochtee formation which consists of argillaceous delomite with interbedded gypsum, anhydrite and shale strata.	8 ft. to 10 ft.		During construction 525 ft. (I.G.L.D.)			Bedrock: For OBE: 150 $KIPS/ft^2$ For SSE: 180 $KIPS/ft^2$	Bedrock: For OBE: 0.01 For SSE: 0.02	
Vol. 1 Sec. 2.5.1.10.2 p. 2-126 to 128	Vol. 1, Sec. 2.5.1.8, p. 2-123 and p. 2-124		Vol. 1 Sec. 2.5.1.7 p. 2-123	Vol. 1 Sec. 2.5.1.5 p. 2-122		Vol. 2 Sec. 3.7.2 p. 3-52 to 55	Vol. 1, Sec. 2.5.1.8, p. 2-124	Vol. 1 Sec. 2.5.1.8 p. 2-124	

SEISMIC REVIEW TABLE

STRUCTURES

DAMPING OBE/SSE (% critical damping)

	$< 1/4\ \sigma_y$	$1/2\sigma_y$	σ_y	$>\sigma_y$
Welded steel	1.0	2.0	5.0	7.0
Bolted and Riveted steel	1.0	5.0	10.0	20.0
Reinforced concrete	1.0	2.0	7.0	10.0

Vol. 2, Table 3-7, pg. 3-50

- D= Dead load of structure and equipment plus other permanent loads, e.g., soil or hydrostatic loads
- L=Live load and piping loads
- R=Force or pressure on structure due to pipe rupture
- T_o=Thermal loads due to temp. gradient, operating
- H_o=Force due to thermal expansion of pipes, operating
- T_a=Thermal loads due to temp. gradient, accident
- H_a=Force on structure due to thermal exp., accident
- E=force due to OBE
- E'=force due to SSE
- W=Wind load–wind velocity 90 mph at 30 ft. above gr.
- W'=Tornado loads including differential pressure

DESIGN CRITERIA

LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Class I Structures: Operation during normal and OBE conditions	
Concrete	**Concrete**
$U=1.5D + 1.8L$	A.C.I. Code. 318-63
$U=1.25(D + L + H_o + E) + 1.0\ T_o$	Ultimate strength method
$U=1.25(D + L + H_o + W) + 1.0\ T_o$	
$U=0.9D + 1.25(H_o + E) + 1.0\ T_o$	
$U=0.9D + 1.25(H_o + W) + 1.0\ T_o$	
Structural steel	**Structural steel**
$D + L$	f_s
$D + L + T_o + H_o + E$	$1.25f_s$
$D + L + T_o + H_o + W$	$1.33f_s$
During accident and SSE conditions:	
Concrete:	
$U=1.0D + 1.0L + 1.25E + 1.0T_a + 1.0H_a + 1.0R$	
$U=1.0D + 1.25E + 1.0T_a + 1.0H_a + 1.0R$	
$U=1.0D + 1.0L + 1.0E' + 1.0T_o + 1.25H_o + 1.0R$	
$U=1.0D + 1.0L + 1.0E' + 1.0T_a + 1.0H_a + 1.0R$	
$U=1.0D + 1.0L + 1.0W' + 1.0T_o + 1.25\ H_o$	
Structural Steel	
$D + L + R + T_o + H_o + E'$	$1.5f_s$
$D + L + R + T_a + H_a + E'$	$1.5f_s$
Vol. 2, Sec. 3.1.1.3, pg. 3-76	Vol. 2, Sec. 3.8.1.1.6, pg. 3-72

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
		DESIGN CRITERIA	
DAMPING OBE/SSE	METHOD OF QUALIFICATION	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
<table><tr><th></th><th>$< 1/4\sigma_y$</th><th>$1/2\sigma_y$</th><th>σ_y</th><th>$> \sigma_y$</th></tr><tr><td>Vital piping</td><td>0.5</td><td>0.5</td><td>2.0</td><td>—</td></tr><tr><td>Piping</td><td>0.5</td><td>1.0</td><td>2.0</td><td>5.0</td></tr></table>	analytical	Code Class I Pressure Vessels **Condition** — **Stress Intensity** Normal: $P_M \le S_M$ P_M (or P_L) + $P_B \le 1.5\ S_M$ P_M (or P_L) + $P_B + Q \le 3.0\ S_M$ P_M (or P_L) + $P_B + Q + F \le S_E$ Emergency: $P_M \le 1.2\ S_M$ or S_Y whichever is larger P_M (or P_L) + $P_B \le 1.5\ (1.2\ S_M)$ or 1.5 S_Y whichever is larger or 0.8 C_L	ASME BPVC, Section III , Class "A" 1968 edition for reactor vessel, steam generator, pressurizer, reactor coolant pump, casing. ANSI B 31.7 - 1968 for piping
Vol. 2, Table 3-7, p. 3-50	Vol. 2 Sec. 3.7.2.1 p. 3-52	See Table 5-13 for upset and faulted condition Tables 5-12,13,14,15,16,17, 18 p. 5-79 through 5-85	Tables 5-10, p. 5-77

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	IEEE 344-1971 and IEEE 336-1971 Vol. 2, Sec. 3.10, p. 3-176, Vol. 2 Append. 3D, p. 3D-85

SEISMIC REVIEW TABLE

Docket Number
50-315, 316

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA: OBE HOR. g	OBE VERT. g	SSE INTENSITY MM	SSE HOR. g	SSE VERT. g	EARTHQUAKE TIME HISTORY	METHOD OF COMBINATION: NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	DESIGN SPECTRA: TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
Donald C. Cook Nuclear Plant Units No. 1 & 2 Reactor type: PWR Containment type: Ice Condenser ([redacted] concrete) NSSS Manufacturer: Westinghouse Architect Engineer: American Electric Power Service Corporation	0.10	0.067	VII	0.20	0.133	El Centro (as presented in TID 7024) Normalized to the recommended ground acceleration was used to develop response spectra.	Two-thirds of horizontal taken as vertical and combined with the full horizontal response. No further details available.	SRSS	Response spectra as shown in Figs. 2.5-2 and 2.5-3 were generated from El Centro earthquake.	Time-history method.
3-69/10-74	Vol. I Sec. 2.8.6 p. 2.8-2	Vol. I Sec. 2.8.6 p. 2.8-2		Vol. I Sec.2.8.6 p. 2.8-2	Vol. I Sec.2.8.6 p. 2.8-2	Vol. I Sec. 2.5.2 p. 2.5-5	AEC Question 5.74 p. 5.74-1	Vol. IX Amend. 9 Q.5.72-1	Vol. I Sec. 2.5.2 p. 2.5-5	Amend. 16 Question 5.71 p. 5.71-6

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Mat foundation Depth not available. Sec. 2.5.2 p. 2.5.2	Compact sand, recompacted sand or stiff clay deposits on shale bedrock. Sec.2.3.2 p. 2.3-4	120 to 200 ft Sec. 2.3 Fig.2.3-2	900 fps Vol. IX, Amend. 19, p. 5.85-2	ground water elevation 593 ft Vol. I, Sec. 2.4.2 p. 2.4-4	Not available.	Stick model with soil springs. Amend. 16. Question 5.71 Fig. 5.71-1	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment structure: (with DBA)	4.0/7.0	For containment: C = (1.0 ± 0.05) D + 1.5 P = 1.0 (T+TL) + 1.0 B C = (1.0 ∓ 0.05) D + 1.25 P + 1.0 (T'+TL') + 1.25 E + 1.0 B C = (1.0 ∓ 0.05) D + 1.0 P + 1.0 (T''+TL'') + 1.0 E' + 1.0 B C = (1.0 ∓ 0.05) D + 1.0 (T'''+TL''') + 1.0 B + 1.0 W' + 1.0 P C = (1.0 ∓ 0.05) D + 1.0 (T'''+TL''') + 1.0 B C = (1.0 ± 0.05) D + 1.15 P	ACI-318-63, Ultimate strength design.
(without DBA)	2.0/5.0		
Welded steel structure	1.0/1.0		
Bolted or rivited steel assemblies	2.0/2.0		
Amend. 9 Question 5.85, p. 5.85-2		Sec. 5.2.2.3 p. 5.2-18	Amendment 9, Question 5.1-1 Appendix B-9

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
	DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping	0.5/0.5	Analytical and Testing	For pressure vessels: 1. (a) $P_m \leq S_m$ (b) P_m (or P_L) + $P_B \leq 1.5\ S_m$ (c) P_m (or P_L) + P_B + Q $\leq 3.0\ S_m$ Normal condition 2. (a) $P_m \leq S_m$ (b) $P_m + P_B \leq 1.5\ S_m$ (c) $P_m + P_B + Q \leq 3.0\ S_m$ Upset condition For pressure piping: 1. (a) $P_m \leq S$ (b) $P_m + P_B \leq S$ Normal condition 2. (a) $P_m \leq 1.2\ S$ (b) $P_m + P_B \leq 1.2\ S$ Upset condition	1. ASME BPVC, Section III 2. USAS.1, B31.1 code (power piping)
Amendment 19, Q. 5.85 p. 5.85-2		Amendment 25 Q. 4.31-1	Tables 1 and 2, p. B-18 and p. B-19.	

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE *

Docket Number
50-010

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Dresden Nuclear Power Station Unit 1 Reactor type: BWR Containment type: Pre-Mark (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel 5-56/9-59	None used (0.10)**	None used (0.067)**	None used	None used (0.20)**	None used (0.13) **	None used	None used Two comps.,** vertical + worst case horizontal	None used SRSS **	None used	No floor response spectra generated UBC, 1955 used for containment (Zone 2) and internal concrete structure (Zone 1) Housner spectra Times 2 used for ECCS and Core Spray System.

* Data are obtained from FHSR Docket 50-010 and SEPB Report "Seismic Design Bases and Criteria for Dresden Unit 1 Nuclear Generating Station," EDAC 175-130.03, January 1979.
** Used for ECCS and Core Spray System only.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Circular concrete foundation 37 ft. below grade.	Limestone Shale Dolomited Limestone	20-45 ft. 70 ft. 100-400 ft.	Not used, bedrock site	"Groundwater found @ various levels beneath the site".	Dresden Dam	No SSI model used	Not used	Not used	Not used

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% Critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	<u>Internal Concrete Structures:</u> E + pressure + equipment (E = 0.025g)	UBC, 1955 ACI, 318-55 AISC, 1955

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE	(% Critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
			LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES	
Vital piping	0.5	None	Containment: 1.) 0.033g 2.) pressure + snow + wind	Steel Containment Sphere and NSSS: ASME Section VIII (1955 ed.) and UBC, 1955	
Welded assembly	1.0		NSSS: 1.) 0.025g 2.) operational transients	Piping and ECCS, Core Spray: ANSI B31.7, and ASME Sec. III, (1974 ed.)	
Bolted assembly	1.0		ECCS: 1.) earthquake + operational + blowdown		

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE*

Docket Number
50-237,249

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Dresden Nuclear Power Station Unit 2 and 3 Reactor type: BWR Containment type: Mark-I (steel) NSSS Manufacturer: General Electric Architect Engineer: Sargent and Lundy Engineers.	0.10	0.067	VII	0.20	0.133	N-S component of the El Centro Earthquake (May, 1940) normalized to a maximum ground acceleration of 0.1g was used for time history analysis.	2 comp., greater horizontal + vertical, absolute method	SRSS (reactor, turbine bldg., and drywell analyzed by time history method)	Housner-(El Centro T-H envelops the Housner spectra except for high frequency end.)	Equipment and piping analyzed by either response spectrum or equivalent static method. Floor response spectra for pressure vessel, isolation condensor, turbine building, control room, etc. are derived by factoring up the Housner Ground Response Spectra to account for the maximum floor acceleration determined from the time history analysis. Static coefficients were also used for APCI and Core Spray Equipment. Floor response spectra from Brown's Ferry used for recirculating Loop Piping, feedwater and mainsteam lines.
Unit 2: 1-66/12-69 Unit 3: 10-66/1-71	p. 12.1-9	p. 12.1-9	p. 12.1-9	p. 12.1-9	p.12.1-9	Question II.A.1 Docket 50237-16 (microfiche)				

*Information was obtained from BNL Docket Search and SEPB Report "Seismic Review of Dresden Unit 2 for the Systematic Evaluation Program", NUREG/CR-0891, July 1979.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION: TYPE	BEARING INFORMATION: THICKNESS	BEARING INFORMATION: V_s PROFILE (calculated)	GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE (calculated)	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
Reinforced concrete mat founded on competent rock p. 12.1-10	The site consists of an upper layer of Pennsyl- vanian Pottsville sandstone of variable thickness which is 40-50 ft. Next below is a layer of a- bout 15 to 35 ft. of Ordovician Maquoketa Divine limestone based on a 65 ft. layer of Maquoketa p. III-1-3	dolomitic shale. The Ordovician system has a total thickness approaching 1000 ft. with the Cambrian system next below. Brecciated rock is found on same cross sections and is indicative of ancient faulting.	Sandstone = 2,600 fps Limestone = 8,600 fps Argillaceous Dolomite = 4,700 fps Shale = 3,900 fps Dolomite Shale = 4,700 fps p. III-2-21	Not available	Dresden Dam p. 2.5-1	Fig. 12.1.8 and Fig. 12.1.9 indicate stick model with fixed base. p. 12.1-12 Lumped mass and stick model (torsional effects not considered.)	Sandstone = 18.7×10^4 psi Limestone = 250×10^4 psi Argillaceous Dolomite = 68×10^4 psi Shale = 44×10^4 psi Dolomite Shale = 74×10^4 psi p. III-2-21	Not available	Not available

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete structures Steel frame structures	5.0 2.0	Reactor building + all other Class I structures a) D + R + E	a) Normal allowable code stresses, AISC for structural steel, ACI-318-63 without increase for seismic
Welded assemblies Bolted and Riveted assemblies	1.0 2.0	b) D + R + E′ Stresses are limited to the minimum yield pt. case an analysis, using the limit-design ap- the energy absorption capacity which should be energy input. AEC publication TID-7024 "Nuclear 5.7.	as a general case. In this proach, is made to determine such that it exceeds the Reactor and Earthquake" Sec.
Reactor and turbine building Ventilation stack	5.0 5.0	Primary containment (including penetrations) a) D + P + H + T + E	a) ASME, Sec. III, Class B, without the usual increase for seismic loadings.
Drywell Control room	5.0 5.0	b) D + P + R + H + T + E	Same as (a), above except local yielding is permitted in the area of jet force where the shell is backed up by concrete. In areas not backed up by concrete, primary local membrane stresses at the jet force <0.9 x yield pt. of material at 300°F.
Amend 13 - Unit 2-SAR Amend 14 - Unit 3-SAR		c) D + P + R + H + T + E′ Primary membrane stresses, in general, of the material. If the total stresses analysis was made to determine that t exceeded the energy input from the ear as in (b), above, is applied to the ef loading condition.	do not exceed the yield pt. exceeded yield pt. an he energy absorption capacity thquake. The same criteria fect of jet force for this

D = Dead load of structure and equipment plus any other permanent loads contributing stress.
P = Pressure due to loss-of-coolant accident, R = Jet force on pressure on structure due to rupture of any one pipe,
H = Force on structure due to thermal expansion of pipes under operation conditions, T = Thermal loads on containment due to loss-of-coolant accident, E = Design earthquake load.

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Suppression chamber	2.0	Analytical model	Reactor Primary Vessel Internals a) D + E	a) ASME, Sec. III Class A vessel
Feedwater lines	0.5		b) D + E′ b) The secondary and primary plus secondary stresses are examined on a rational basis taking into account elastic and plastic strains. These strains are limit to preclude failure by deformation which would compromised any of the engineered safeguards or prevent safe shut-down of the reactor.	
Vital piping systems	0.5			
Reactor pressure vessel	2.0			
Recirculation loop piping	0.5		c) P + D	c) ASME, Sec. III, Class A
Main steam lines	0.5		Reactor Primary Vessel Supports a) D + H + E	a) AISC for structural steel ACI for reinforced concrete
Suppression chamber ring header	0.5		b) D + H + R + E	b) Stresses do not exceed: - 150% of AISC allowable for structural steel - 90% of yield stress for reinforcing bars - 85% of ultimate stress for concrete
		Question 2.16 Amend. 7,8	c) D + H + E′ p. 12. 1-6	c) The design is such that energy absorption capacity exceeds energy input.

ECCS: a.) D + T + H + E — a.) Piping - ASA B 31.1 (1955 ed.) and code cases
Pumps - ASME Sec. III, Class C
Shellside - ASME Sec. III, Class C and TEMA C
Tubeside - ASME Sec. VIII, TEMA C

b.) D + T + H + E′ — b.) Same as P + D above

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Analysis and Generic Testing	Battery racks - No structural design calculations Instrumentation and control room panels - GE generic tests* Motor Control Center - Cutler Hammer Co. Generic Tests ** - Vibration test and analysis of 7700 Line Motor Control Center, # 70ICS100, 8-70 Transformers - No tests or calculations Cable trays - S. and L. Engrs., Specs, for Cable Pans and Hangers, Spec. K-2197	Not available

* GE - "Seismic Testing of Instrumentation" Dresden 2, 1-71
** Wyle Labs - "Seismic Simulation Test Report for Modified Unitrol Motor Control Center, Report 43746-1, 10-77

SEISMIC REVIEW TABLE

Docket Number
50-331

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Duane Arnold Energy Center Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel	For structures on bedrock or loft fill: 0.06 For structure on 30-50 ft. of soil: 0.09.	For structure on bedrock: 0.05 For structure on soil: 0.06	Not available.	For structures on bedrock or 10 ft. of fill: 0.12 For structure on 30-50 ft. of soil: 0.18	Structure on rock: 0.10 Structure on 30-50 ft. of soil: 0.12	1. 1935 Helena, Montana earthquake. 2. 1952 Taft, California earthquake.	The earthquake conditions were applied to the structure in the direction of each of their principal axes.	Direct addition (Time history) SRSS (Spectrum analysis)	Response spectra developed for stuctures on: (1) Bedrock: 1935 Helena, Montana earthquake, (2) Compact fill and/or soil overlying bedrock: 1952 Taft, California earthquake.	Time history method using developed earthquake time history.
6-70/2-74	Sec. 2.6.2.1.1 p. 2.6-24 Table 2.6-2	Sec. 2.6.2.1.1 p. 2.6-24 Table 2.6-2		Sec. 2.6.2.1.1 p. 2.6-40 Table 2.6-3	Sec. 2.6.2.5.3 p. 2.6-40	Sec. 2.6.2.5.3 p. 2.6-40	Sec. C.5.2.3.1 p. C.5-5	p. C.5-5 p. C.5-13	Sec. 2.6.2.5.3 p. 2.6-40	Sec. C.5.2.3.1 p. C.5-6

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reactor building: mat foundation on bedrock. Depth: not available.	Surficial deposits of clayey silt, sand, and gravel. Glacial till. Wapsipinicon formation (limestone and dolomite.	25 feet to more than 100 feet thick. About 67 feet thick.	V_s value computed: Surficial deposit: 500 fps Glacial till: 1800 fps. Limestone: 8600 fps.	About 8 feet below the existing ground surface.	"There are 12 low head dams."	Figure C.5-5 indicates stick model with soil springs.	Alluvial sand: 0.5×10^6 psf Glacial till: 0.7×10^6 psf Rock: 200×10^6 psf	Foundations, rock, soil: OBE and DBE: 5.0% of critical damping.	Not available.
Sec. 2.6.3.1.1 p. 2.6-46	Sec. 2.6.1.1.1 p. 2.6-1 Fig. 2.6-9	Sec. 2.6.1.1.1 p. 2.6-1 Fig. 2.6-9	Fig. 2.6-9	Fig. 2.6-12	Sec. 2.5.1 p. 2.5-1,2	Sec. C.5.2.3.1 p. C.5-5	Table 2.6-4 p. 2.6-80	Table C.5-1	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment structure and all internal concrete structures:	2.0/5.0	(1) Normal loads + operating basis earthquake (2) Normal loads + maximum probable flood (3) Normal loads + design basis earthquake (4) Normal loads + tornado loads (5) Normal loads + design basis loss-of-coolant accident reference For further information refer to Sec. 12.4.2, p. 12.4-1.	ACI-318-63 Ultimate strength design.
Other conventionally reinforced concrete structures, such as shear walls or rigid frames:	5.0/5.0		
Table C.5-1		p. 12.4-3	p. 12.4-7

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE	(% of critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
			LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES	
Welded structural steel assemblies:	1.0/1.0	Analytical and testing.	Table C.2-1 (partial) Summary of Loading Conditions and Criteria Reactor Pressure Vessel - Normal - ASME Code, Special Criteria Upset - ASME Code, Special Criteria (Table C.2-2) Emergency - ASME Code, Special Criteria (Table C.2-2) Faulted - ASME Code, Special Criteria (Table C.2-2) Piping - Normal - Industry Codes, Table C.2-2 Upset - Industry Codes, Table C.2-2 Emergency - Industry Codes, Table C.2-2 Faulted - Industry Codes, Table C.2-2	ANSI B31.1.0-1967 B31.7 ASME, BPVC	
Bolted or riveted steel assemblies:	2.0/2.0				
Piping systems:	0.5/1.0				
Table C.5-1		Sec. C.5.2.3-1 p. C.5-6,7	Tables C.2-1 through C.2-25,p. C.2-11 through C.2-73	Sec. A.1.2 p. A.1-3	

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analysis or testing.	<u>GE equipment</u>: "All instrumentation required for nuclear safety is capable of performing all functions important to safety during normal operation, during DBA and post-accident operation. Qualification is achieved by test and/or analysis at acceleration values of 1.5g horizontal and 0.5g vertical over a frequency of 0.25 to 33 Hz". <u>Bechtel supplied equipment</u>: "Purchase specifications will require that each type of Class 1 device be individually qualified by vibration test or suitable analysis. The methods ... will meet the general requirements of IEEE Standard 344-1971.	IEEE 344-1971
	Sec. C.5.2 .3.1 p. C.5-6, 7	For futher information refer to: Appendix M: Section M.3.3, p. M.3-27 through p. M.3-34	

SEISMIC REVIEW TABLE

Docket Number
50-321

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Edwin I. Hatch Nuclear Power Plant Unit No. 1 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel	0.08	0.053	VII	0.15	0.10	N-S component of 1940 El-Centro earthquake.	2 components: Worst horizontal component plus vertical combined simultaneously	SRSS including closely spaced modes.	Conform to the average spectra by G.W. Housner for T $\leq$4 s. Normalized to the peaks (horizontal) of OBE and SSE.	Time-history method Class II UBC
9-69/8-74	Sec. 12.3.3.2 p. 12-8		Sec. 2.5.9 p. 2-33	Sec. 12.3.3.2 p. 12-8	Sec. 12.3.3.2 p. 12-8	Sec. 12.6.2.1 p. 12-21	p. C-13	Sec. 12.6.2.1 p. 12-20	Sec. 2.5.9 p. 2-33 Fig. 2.5-5 and 6	Sec. 12.6.2.1 p. 12-21

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat foundations for the following buildings: reactor, turbine, control, diesel generator, and radwaste. The foundation for the main stack is a reinforced concrete mat on steel H-piles.	Duplin: (cemented clay-sand grading to sandy clay).	135 ft	2450 fps	Summary of domestic well study is given in Table 2.4-3, pp. 2-18 and 2-19 of Section 2.4.6.2. Summary of Piezometer Installation Data is given in Table 2.4-4, pp. 2-20 and 2-21 of Section 2.4.6.2 No liquefaction potential has been found.	Not available.	Stick model with soil springs.	23,300 ksf	Translation and rotation of foundation soil - 4.5%OBE - 5.5%DBE	Unclear information Ref: PSAR Sec. XII-31
	Beneath: (sand, sandy-clay)	10 to 70 ft							
	Clay, sand, gravel, etc.	65 ft							
	Crystalline basement rock.	4000 ft							
							Amendment 14, 4/72 Vol. VIII of FSAR Table Q 12.3.3.2.4-1 of Question 12.3.3.2.4		
Sec. 12.5 p. 12-18	Sec. 2.7.4 p. 2-41		Amend. 14 (4/72) p. 12.3.3.2.4-2	Sec. 2.7.7 p. 2-45		Amendment 12 12/72 Sec. 12.6.2.1 p. 12-20 Fig. 12.6-1		Table 12.3-2 p. 12-10	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete structures:	3.0/5.0	Class I structures 1. Primary containment. (a) D+L+H+T+E (b) D+L+H+P+R+T+E (c) D+L+H+P+R+T+E' (d) D+E+F 2. Reactor pressure vessel support. (a) D+L+H+E (b) D+L+H+R+P+T (c) D+L+H+T+P+T+E (c) D+L+H+R+P+T+E' 3. Reactor building and all other Class I structures. (a) D+L+H+E (b) D+L+H+W (c) D+L+H+E' (d) D+L+H+W' 4. Reactor building crane structure. (a) D+L+C+I (b) D+L+C+E (c) D+L+C+E' (d) D+L+C+W (d) D+L+C+W' Class II structures: designed according to applicable codes and standards. NOTE: D = dead load, L = live load, C = crane load, I = impact load, P = pressure due to LOCA, R = jet force, T = thermal load, E = OBE, E' = SSE, W = wind, W' = tornado wind, and F = hydrostatic. Sec. 12.4, p. 12-15	They are classified according to the load combination case. For details, see Sec. 12.4, pp. 12-15 and 12-16. Generally used: ASME, Sec. III, Class B. For steel structures, AISC. For concrete structure: ACI 318-63 and 307-69
Steel frame structures:	3.0/5.0		
Bolted and riveted assemblies:	3.0/5.0		
Welded assemblies:	2.0/3.0		
Vital piping:	0.5/1.0		
Translation and rotation of foundation soil:	4.5/5.5		
Amendment 12, 2/72, Vol. III Sec. 12, Table 12.3-2 p. 12-10			

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping: Sec. A.3.1.4 p. A-4	0.5/2.0	Analytical and testing. Amendment 13 3/72 Sections C.1.1 and C.12, p. C-1	Reactor vessel: 1. Normal + OBE 2. Normal + piping rupture or normal + SSE 3. Normal + SSE + piping rupture Piping: Dead loads + external loads + thermal loads. 1. Dead + pressure 2. Dead + pressure + OBE 3. Dead + pressure + thermal } USAS B31.1.10 4. Dead + pressure + SSE 5. Dead + maximum pressure + OBE 6. Dead + maximum pressure + SSE } Low probability of occurrence More details on Table C-3.1 of Section: NSSS Equipment Loading Design on FSAR, Vol. IV, pp. C-14 to C-46.	ASME, BPVC, Section III, Nuclear Vessels, 1965 Edition and Winter 1966 Addenda with additions listed on page I-1 of Appendix I of Reactor Pressure Vessel Report. pp. C-10, C-12

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-366

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Edwin I. Hatch Nuclear Power Plant Unit No. 2 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel	0.08	0.053	VII	0.15	0.10	Modified Taft 1952 horizontal component was used for developing synthetic acceleration time history.	3 components: Each horizontal combined with the vertical simultaeously, resulting in two separate seismic cases.	SRSS with close modes summed absolutely.	Modified Newmark design spectra.	Time-history method.
9-69/8-74	Sec. 2.5.2.11 p. 25-26	Sec. 2.5.2.11 p. 25-26	Sec. 2.5.2.10 p. 25	Sec. 2.5.2.10 p. 25	Sec. 2.5.2.10 p. 25	Sec. 3.7A.1.2 p. 3.7A-1	Sec. 3.7A.3.7 Sec. 3.7B.3.7 p.9	Sec. 3.7A.2.1.1 Sec. 3.7A.2.2 Sec. 3.7A.3.7	Sec. 3.7A.1.1 Figs. 3.7A1-3.7A6	Sec. 3.7B.2.6 Sec. 3.7B.2.3 Sec. 3.7B.2.8

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat 27'2" thick at middle dry well and 12'4" thick at other sections.	Major and minor struct: Upper Miocene Dublin locally cemented sand to sandy clay Upon Hawthorne sandy clay.	To a depth of 135' (ft). Below Dublin down 10'-70' (ft).	2450 ± 200 fps	el.70 to el.75 ft.	2 upstream of plant, Caltamaha River Basin 1) Sinclair Dam on Oconee Riv. 2) Lloyd Shoals Dam, Ocmulgee River.	Stickmodel with soil springs	Not available.	Not available.	Not available.
Sec. 3.8.5.1b p. 3.8-76 Fig. 3.8-31 & 32	Sec. 2.5.2.1 p. 23 Sec. 2A. p. 4 Figures 2A-2 thru 2A-3EE	Sec. 2.5.2.1 p. 23 Sec. 2A.2 p. 4 Figures 2A-2 thru 2A-3EE	Sec. 2A.1.4 p. 2A.1-3 Fig. 2A-5 and 2A-6	Sec. 2.5.4.6 p. 2.5-30		Sec. 3.7A.2.4 Sec. 3.7A.2.5 p. 5			

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete structure:	3.0/5.0	Steel containment (a) Initial and final testings	ASME, BPVC, Sec. III
Steel frame structures:	3.0/5.0	(1) $D+L+P_t+T_t+E$	AISC 1969 Ed.
Bolted and riveted assemblies:	3.0/5.0	(2) $D+L+P_t+T_t+E'$	ACI 318-63
Welded assemblies:	2.0/3.0	(b) Normal operating (1) $D+L+T_o+R_o+E$	
Translation and rotation of soil: (NSSS)-	4.0/5.0	(2) $D+L+T_o+R_o+E'$ (3) $D+L+T_e+R_e+P_e+E$	
Drywell-building (coupled):	3.0/5.0	(4) $D+L+T_e+R_e+P_e+E'$	
Suppression chamber:	2.0/3.0	(c) Refueling (1) $D+L+E$	
Reactor pressure vessel, support skirt, shroud head, separator and guide tubes:	2.0/3.0	(2) $D+L+E'$ (d) Accident (1) $D+L+T_a+R_a+P_a+E$ (2) $D+L+T_a+R_a+P_a+E'$	
Fuel:	7.0/7.0	(3) $D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E'$ (e) Flood (1) $D+L+E+F$ Sec. 3.8.2.3 p. 7 Sec. 3.8.3.3 p. 47 Sec. 3.8.4.3 p. 58 Sec. 3.8.5.3 p. 78	Sec. 3.8.2.2 p. 4 Sec. 3.8.3.2 p. 45 Sec. 3.8.4.2 p. 57 Sec. 3.8.5.2 p. 78
Table 3.7A-1 and 3.7B-1			

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
			LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES	
Vital piping systems Fuel	0.5/1.0 7.0/7.0	Analytical and supplementary testing (NSSS) Secs. 3.7B.2.1 3.7B.2.1.6.1 3.7B.2.1.6.2 3.7B.2.1.7.1 3.7B.2.1.7.2 3.7B.2.1.8	Load combination definitions are according to ASME Sec III NB-3200 through NB-3600. For details see tables below, e.g., Table 3.9-4, "Reactor Pressure Vessel Internals and Associated Piping." Table 3.9-4, through 3.9-64	ASME, BPVC, Section III Table 3.9.-1, 3.9-2 Sec. 3.9.1.6 p. 3.9-8	

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and testing. Secs. 3.7A, A.3.1 3.7A, A.3.2 Table 3.9-23	Not available.	Seismic class I electrical equipment IEEE Std. 344-1971 p. 3.7A.A-1 to 3.7A.A-6 Tubing- ASME BPVC Section III

SEISMIC REVIEW TABLE

Docket Number
50-285

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM.	HOR. g	VERT. g					
Fort Calhoun Station Unit #1 Reactor type: PWR Containment type: Without Buttresses (Prestressed Concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Gibbs & Hill, Inc.	0.08	.053	Unclear information	0.17	.0113	Time history—1940 El Centro and 1952 Taft normalized to the ground acceleration of the maximum hypothetical earthquake are used for developing floor response spectra.	3 components. Combination not available.	SRSS	Response spectra conform to the average spectra developed by Housner for frequency > 0.33 HZ and Newmark for frequency < 0.33 HZ.	Time history method.
6-68/5-73	Sec. 2.4 p. 2.4-3	Sec. 2.4 p. 2.4-3	Sec. 2.4 p. 2.4.1	Sec. 2.4 p. 2.4.3	Sec. 2.4 p. 2.4.3	Sec. F.2.2.4 p. F-10	App. F Sec. F.2.5 p. F-12	App. F Sec. F.2.2.3 p. F-9	App. F Sec. F.2.1.4 p. F-6	App. F Sec. F.2 p. F.10 & F.14

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
High strength concrete mat supported by pile foundation resting on bedrock (containment, auxiliary bldg.)	Compact granular. Fluvial deposits on lime-stone. Bedrock underlain by rock strata.	60 ft 4-8 ft 19-21 ft	Not available.	Missouri River Valley. Domestic wells depth 20 ft to 35 ft. Commercial wells depth 50 ft to 75 ft.	Gavin Point Fort Randall Big Bend Oahe Garrison Fort Peck	Stick model with soil springs.	Not available.	Not available.	0.05 SSE 0.02 OBE
Sec. 5.1 p. 5.1.1 Covering letter "Dames & Moore" App. C p. 10	Sec. 5.1 p. 5.1.1 App. C p. 6	Sec. 5.1 p. 5.1.1 App. C p. 6		Sec. 2.7.2 p. 2.7-6	Sec. 2.7 p. 2.7-1	Sec. F.2.2.3 p. F-8			Sec. F.2.2.3 p. F.9

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment structure: Concrete support structures for reactor vessel and steam generators: Steel Assemblies: Bolted or riveted Welded Vital piping systems: Rigid vault type concrete structures: Framed concrete structures: Sec. F-2.1.3 p. F-6	2.0/2.0 2.0/2.0 2.0/2.0 1.0/1.0 0.5/0.5 2.0/5.0 5.0/7.0	1. D+L+S+T''' 2. D+L+S+T'''+W or E 3. D+L+P+S+T+W or E where: D = Dead load including equipment weights and hydrostatic loading L = Live load S = Post-tensioning load (which varies with time) P = Accident design pressure T = Thermal loads based on a temperature corresponding to pressure P W = Wind load E = Design earthquake T'''= Thermal loads based on normal operating temperature For further details refer to section 5.5. Sec. 5.5 p. 5.5-1 to 5.5-5a	Ultimate strength method ACI 318-63 Modified ultimate strength design No loss of function design for extreme environmental loading Sec. F.2.1.1 p. 5.5-1 Sec. 5.5 p. F.3

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Mechanical equipment: Piping: Appendix F Sec. F.2.1.3 p. F.6 Table F.2	2.0/2.0 0.5/0.5	Analytical and testing. Appendix F Sec. F.2.2.2 p. F-7C	Reactor vessel: 1. Design loading + OBE $P_m \leq S_m$, $P_b+P_L \leq 1.5S_m$ 2. Normal operation + SSE $P_m \leq S_d$, $P_b \leq 1.5\left[1-\left(\frac{P_m}{S_d}\right)^2\right] S_d$ 3. Normal operation + SSE + pipe rupture $P_m \leq S_L$, $P_b \leq 1.5\left[1-\left(\frac{P_m}{S_L}\right)^2\right] S_L$ where $S_L = S_y + (1/3)(S_u - S_y)$ $S_d = 1.2S_m$ Piping: 1. Design load + OBE Applicable code allowable 2. N.O. + SSE $P_m \leq S_d$, $P_b \leq 4/\pi\ S_d \cos\left(\frac{\pi}{2}\cdot\frac{P_m}{S_d}\right)$ 3. N.O. + SSE + pipe rupture $P_m \leq S_L$, $P_b \leq 4/\pi\ S_L \cos\left(\frac{\pi}{2}\cdot\frac{P_m}{S_L}\right)$ For reactor vessel and piping Sec. F.2.1.2 Table F.1, p. F.4 and F.5, Appendix F	ASME, BPVC, Section III USAS, B31.1 and B31.7 Appendix F Sec. F.2.1.1 p. F.3

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Shop test, prototype test, field test or seismic analysis to meet Class I seismic criteria. Appendix F Sec. 6.14 Sec. F.2.2.2 p. 6.1-4 p. F.7.C, 7d	"Special seismic restraints will be installed at the electrical cable trays. The cable will be supported vertically and horizontally so as to meet the stress criteria under all conditions including postulated earthquakes." Sec. F.2.2.2 p. F.7.C	According to IEEE 344 "Guide for Seismic Qualification of Class I Equipment for Nuclear Power Generating Station" Sec. F.2.2.2 and Sec. 7.2.2 p. F.7.C and p. 7.2.1

SEISMIC REVIEW TABLE

Docket Number
50-267

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Fort St. Vrain Nuclear Generating Station (Unit 1) Reactor type: HTGR Containment type: Prestressed Concrete NSSS Manufacturer: Gulf General Atomic Architect Engineer: Sargent and Lundy Engineers 9-68/12-73	0.05 Amend. 14 Sec. 5.2.1.1 p. 5.2-4	0.033 Amend. 14 Sec. 5.2.1. p. 5.2-4	VII	0.10 Amend. 14 Sec. 5.2.1.1 p. 5.2-4	0.067 Amend. 14 Sec. 5.2.1.1 p. 5.2-4	TID-7024, "Nuclear Reactors and Earthquakes",AEC, 8/63 p. 14.1-1, App. E.13	The horizontal results from spectral analysis were combined simultaneously with the vertical p. 5.3-33	PCRV: "Linear superposition of all modal contributions". All other Cat. I struct: SRSS p. 14.1-4	Response spectra were developed as recommended in AEC TID-7024. Housner Fig. 14.1-1 Sec. 14.1 p. 14.1-1, 14.1-3	TID-7024 p. 14.1-1 App. E.13

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
1. Reactor, turbine buildings and heavy equipment, as well as the main and service water cooling towers. Straight shaft piers. drilled into the claystone bedrocks. 2. Miscellaneous light equipment. Spread footings. Sec. 2.6 p. 2.6-20	The major plant facilities will be founded on Pierre Shale bedrock (dark gray, silty Above it lies St. Vrain Platte River alluvia sands and gravel	shale). 44 to 54 ft. p 1.2-2 p. 1.2-2	Not available	Ground water level was well below proposed foundation level, except reactor building which extends below the water level. Sec. 2.6 p. 2.6-21	V_S = 1200 fps @ 20 ft. V_S = 2400 fps @ 65 ft. Boring UH1 Table 3-1	Lumped mass model with soil springs p. E. 37-12 Fig. E.13-1	G_S = 850 psi @ 20 ft. G_S = 104,000 psi @ 65 ft. Boring UH1 Table 3-1 p. 3-8	Not available	Not available

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE (% Critical damping)		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete:	2.0/5.0	PCRV: DL + 1.23 NWP + E′ + TL DL + 1.23 NWP + 1.5 TL NWP = Normal working pressure DL = Dead load E′ = SSE earthquake loads TL = Temperature loads	For reactor core support structure: Concrete. ACI 318-63 Metal. ASME B and PV Code Sec. III. Class A Stress Criteria: Operating Principal Comp. $0.45\ Cf'_c$ Principal tension $3\sqrt{f'_c}$ Bearing tendon area $0.6f'_c\ 3\sqrt{ab'/ab'}\ < f'_c$ Bearing: Shear Anchors $0.6f'_c$ average
PCRV. (prestressed concrete reactor vessel)	2.0/5.0		
Welded steel	2.0/5.0		
Bolted steel	2.0/10.0		
Amend. 16, p. 14.1-3		Table E.1-1	Table E.1-1 Sec. 3.2, p. 3.2-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital steel piping Amend. 16, p. 14.1-3	0.5/0.5	Dynamic seismic method for Class I System and piping based on Fig. "SSE" ground acceleration Sec. 1.4 p. 1.4-3 and Tests for Class I systems, Q. 5.1, Amend. 16 Attachment A, p. 5.1-1 and Q. 5.11, Amend. 16 Attachment A p. 5.11-1 and Amend. 1 Attachemnt A p. 5.11-9	For PCRV Internal Steel Structure: a) D. L. + Operating mechanical load $\leq$ 0.667 F_y b) D. L. + Operating mechanical load + Design seismic loads $\leq$ F_y c) D. L. + Operating mechanical + twice design seismic load $\leq$ No loss of safety function Amend. 16, p. 5.21-1, Table 3.2-1	For all piping systems: ANSI B.31.1.0-1967. For containment tank: ASME Code Sec. III-C For coolers: ASME Code Sec VIII Sec. 4.2, p. 4.2-10 Sec. 4.2, p.4.2-28 Sec. 4.2, p.4.2-35

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Tests and inspections. For auxiliary electrical system. Amend. 25 p. 8.4-1	Not available	Not available

SEISMIC REVIEW TABLE*

Docket Number
50-213

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Haddam Neck Nuclear Power Plant, Unit 1. Reactor type: PWR Containment type: Reinforced Concrete Cylindrical NSSS Manufacturer: Westinghouse Architect Engineer: Stone and Webster Engineering Corp. 5-64/6-67	Not used	Not used	Not used	0.17 (For safety related structures, e.g., reactor vessel, reactor coolants system safety injection system, spent fuel storage pit). 0.03 (for non-safety related). 12.5-4	0.11 12.5-4	Not used	For the reactor containment a vertical acceleration component equal to 2/3 the horizontal was assumed to act non-concurrently. Resulting stresses were lower than those from the horizontal component and so vertical accelerations were neglected.	No modal combinations - reactor containment analyzed as one degree of freedom system. For complex systems, the maximum acceleration from the damped Housner curve was used, otherwise primary period value was used. 2.5-4	Housner (JEMD, ASME, Oct. 1959) Fig. 2.5-1	No floor response spectra generated. Housner's "Average Acceleration spectrum" was used for all elevations.

*Information obtained from BNL Docket search and SEPB Report prepared by LLL, EDAL Report # 175-130.01, January 1979.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Containment- 9 ft. mat. Spent fuel pit founded on bedrock with lowerside walls embedded in rock and earth. Major structures are founded directly on the granitic gneiss bedrock. Minor structures are founded either on rock on piles drived to rock or on spread footings in compacted granular fill.	SAMPLE: Boring Loose loam Firm fine sand and gravel boulder schist	 L-11 EL+7.0 to + 5.0 +5.0 to -2.0 -2.0 to -8.0 -8.0 to -30.0	Not available	21 ft. MSL is yard grade. Calculated site flood stage is 15.1 MSL GWL: - 8 ft. MSL	Not available	Fixed base with single degree of freedom (containment).	Not available	Not used	Not used
2.4-2	Fig. 2.4-4			2.3-3					

SEISMIC REVIEW TABLE

STRUCTURES		
DAMPING OBE/SSE (% critical damping)	DESIGN CRITERIA	
	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
R/C containment: Include mat 7.0 R/C framed structure 5.0 Steel framed structures, include support. structure and foundation bolted 2.5 welded 1.0 Table 2.5-2	<u>Reinforcing steel - primary plus secondary</u> operating + incident - 33.3 ksi operating + .03g hor. - 26.7 ksi operating + .03g hor. + incident - 33.3 ksi operating + incident + 0.17g hor. - 40.0 ksi - wind loads up to 150 mph - 30 psf snow and ice (not included in combination) p. 3.2-2 <u>Non-safety related systems</u>: E (=Ø.Ø3g): No loss of function	ACI and ASME Codes plus Rayleigh method and equivalent static loads for seismic p. 3.2-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
		LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping:	Analytical	Reactor coolant Safety Injection System:	Component	Design Code
Carbon steel 0.5		Operating loads + E < working Stress (E = 0.17g)	Steam generator-	ASME Section VIII (1956 ed.)
Stainless steel 1.0			Reactor Coolant Pumps-	ASME Section VIII (1956 ed.)
			Reactor Coolant Piping -	ASA B31.1 (1955 ed.)
			Pressurizer	ASME Section VIII (1956 ed.) and Code Case Nos. 1224 and 1234
Reactor internals and CRD		Main Steam Piping:	Safety and Relief Valves	ASME Section I (1956 ed.) and Code Case Nos. 1224 and 1234
welded 1.0		Operating loads + E < Working Stress (E= 0.03g)		
bolted 2.0			Loop Stop Valves	ASA B16.5 (1957 ed.)
			Loop Check Valves	ASA B16.5 (1957 ed.)
			Pressure Control and Relief System Piping	ASA B31.1 (1955 ed.)
Mechanical equipment includes pumps and fans 2.0			Low Pressure Surge Tank	ASME Section VIII (1956 ed.)

Table 2.5.2

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT

DAMPING OBE/SSE (% Critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
ot available	No testing	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-261

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
H. B. Robinson Nuclear Steam Electric Plant Unit No. 2 Reactor type: PWR Containment type: without buttresses (prestressed concrete) NSSS Manufactuer: Westinghouse Architect Engineer: Ebasco	0.10	0.067	VII	0.20	0.133	Not used.	X and Y (vertical) or Z and Y (vertical) applied together. Combination not available.	Absolute sum.	Housner spectra.	No floor response spectra generated. Housner spectra used for components.
4-67/8-70	p. 5.1.2-6	p. 5.1.2-6		p. 5.1.2-6	p. 5.1.2-6	p. 5A-4	Question III A 11	Question IIA	Figures 2.9-2 9.9-3 p. 2.9-9	p. 5A-4

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
A 144 ft. diameter circular reinforced concrete slab 10 ft. in thickness supported by 923 steel pile. p. 5.1.2-20 TYPE (cont.) over 430 ft. middendorf formations. Sec. 2.8.3 p. 2.8-6 Dock. 50261-104	The piedmont crystalline basement rock at the site is overlaid with 460 ft. of unconsolidated coastal plain sediment. These sediments are comprised: 30 ft. of surface alluvium * (cont.)	The middendorf is made up of sands, silty and sandy clay, sandstone and mudstone. Fig. 2.8-2 Basement Rock Middendorf 430ft. Alluvium 30ft.	Not available.	Not available.	Earth dam at the site has a central vertical clay core and supporting shells of compacted sand. The crest of the dam is at El. 230, the normal pool is at El. 220 and the dam has a maximum height of 50 ft. The crown width of dam is 15 ft. and side slopes are 1(vertical): **	Not available. **DAM (cont.) 3(horizontal) on upstream side and 1 (vertical): 2.5(Horizontal) on downstream with 15 ft. berm at El. 200. Sec. 2.9.8 p. 2.9-10 Dock. 50261-104	Not available.	Not available.	The modal analysis was performed utilizing the same damping factor for each mode. Question III A4

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment structure: Concrete support structure of reactor vessel: Concrete structures above ground: (a) Shear wall (b) Rigid frame Table 5A.1-1 p. 5A-5	2.0 2.0 5.0 5.0	For containment structure: (a) $C=1.0D\pm0.05D+1.5P+1.0(T+TL)+1.0B$ (b) $C=1.0D\pm0.05D+1.25P+1.0(T'+TL')+1.25E+1.0B$ (c) $C=1.0D\pm0.05D+1.0P+1.0(T''+TL'')+1.0E'+1.0B$ (d) $C=1.0D\pm0.05D+1.0P_T+1.0(T_T+TL_o)+1.25WT+1.0B$ (e) $C=1.0D\pm0.05D+1.15P_D$ Symbols used in these formulas are defined on p. 5.1.2-9. p. 5.1.2-8	For containment structure using ACI 318-63 Ultimate strength design.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="7">MECHANICAL & PIPING</th></tr>
<tr><th colspan="2" rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th rowspan="2">METHOD OF QUALIFICATION</th><th colspan="4">DESIGN CRITERIA</th></tr>
<tr><th colspan="3">LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>Vital pipe systems:</td><td>0.5</td><td rowspan="6">Analytical</td><td></td><td>Vessels</td><td>Piping</td><td rowspan="6">Pressure piping:
USAS B31.1

Pressure vessel:
ASME,BPVC

p. 5A-3</td></tr>
<tr><td>Steel assemblies:
(a) Bolted or riveted
(b) Welded</td><td>
2.5
1.0</td><td>1. Normal loads</td><td>$P_m \le S_m$
$P_L+P_B \le 1.5S_m$</td><td>$P_m \le S$
$P_L+P_B \le S$</td></tr>
<tr><td></td><td></td><td>2. Normal + design earthquake loads</td><td>$P_m \le S_m$
$P_L+P_B \le 1.5S_m$</td><td>$P_m \le 1.2S$
$P_L+P_B \le 1.2S$</td></tr>
<tr><td></td><td></td><td>3. Normal + assumed hypothetical earthquake loads</td><td>$P_m \le 1.2S_m$
$P_L+P_B \le 1.2(1.5S_m)$</td><td>$P_m \le 1.2S$
$P_L+P_B \le 1.2(1.5S)$</td></tr>
<tr><td></td><td></td><td>4. Normal + pipe rupture loads</td><td>$P_m \le 1.2S_m$
$P_L+P_B \le 1.2(1.5S_m)$</td><td>$P_m \le 1.2S$
$P_L+P_B \le 1.2(1.5S)$</td></tr>
<tr><td colspan="2">Table 5A.1-1
p. 5A-5</td><td colspan="3">P_m = primary general membrane stress; or stress intensity.
P_L = primary local membrane stress; or stress intensity.
P_B = primary bending stress; or stress intensity.
S_m = stress intensity value from ASME, BPVC Code, Section III
S = allowable stress from USAS B31.1 Code for pressure piping.

Table 5A.3-1</td></tr>
</table>

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Circuits and equipment were subjected to vibration tests which simulated the seismic conditions for the "low seismic" class of plants. p. 7.5-13	Not available.	Electrical equipment: WCAP 7397-L p. 7.5-14 Amendment 10

SEISMIC REVIEW TABLE

Docket Number
50-133

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Humboldt Bay Power Plant, Unit 3 Reactor type: BWR Containment type: Pre-mark (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel	0.25	0.17	VIII	0.50	0.333	Time-histories given in BC-TOP-4A	BC-TOP-4A	BC-TOP-4A	Reg. Guide 1.60, Rev. 1, 1973	Time history BC-TOP-4A
11-60/8-62	p. 1-1	p. 1-1	FHSR Amend. 11 p. 125	FHSR, Amend. 11, p. 162		p. 5-3				p. 5-1

Information gathered from FHSR Amend. 11 (50133-1), Amend. 13 (50133-3)
FSAR Supp. (50133-59), FSAR proposed Amend (50133-124), FSAR Supp. Emergency Plant (50133-183)
and Summary Report of Seismic Design Review, Rev. 3, 1977.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Not available	Sand and alluvium overlaying strata of Hookton and Carlotta formation which are more or less consolidated sands. Gravels and clays and conclomerates with good structural properties.	Not available	Not available	Not available	Not available	2 dimensional finite element model which includes embedded reactor caissions p. 5-1		BC -	TOP 4A

FHSR, Amend 11,
Sec. I, p. 155

SEISMIC REVIEW TABLE

STRUCTURES		
DAMPING OBE/SSE	DESIGN CRITERIA	
	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
R. G. 1.61 (BC-TOP-4A)	<u>Accident Condition</u> Concrete structures: $U = D + L + T_A + H_A + R + 1.5\ P$ $U = D + L + T_A + H_A + R + 1.25\ P + 1.25\ E$ $U = D + L + T_A + H_A + R + P + E'$ $U = D + L + T_o + H_o + E'$ Steel Structures <u>Elastic</u> working stress $1.6S = D + L + T_A + H_A + R + P$ $1.6S = D + L + T_A + H_A + R + P + E$ $1.6S = D + L + T_A + H_A + R + P + E'$ <u>Plastic</u> $0.9\ Y = D + L + T_A + H_A + R + 1.5\ P$ $0.9\ Y = D + L + T_A + H_A + R + 1.25\ P + 1.25\ E$ $0.9\ Y = D + L + T_A + H_A + R + P + E'$ App. B-3	AWS D1.1-74 welded steel tanks for oil storage, API 650, 1973 BC-TOP-9A, Design of structures for missile impact, Rev. 2, 1974 UBC - 1973 ACI -214 - 65 ACI -318 - 71 AISC - 1969 p. C-1 p. C-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA**				
		LOAD COMBINATION				**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Reg. Guide 1.61 (BC-TOP-4A)	Test or Analysis	Piping System				BN-TOP-2, Design for pipe break effects
		Plant Operating Condition	Loading Condition	ASME SEC.III Ref.	Allowable Stress	
		Normal	P + W	Eq.(8) of NC-3652.1	S_H	
		Upset	P + W + OBE P + W + FV*	Eq.(9) of NC-3652.2	$1.2\ S_H$	
		Faulted	P + W + SSE	Eq.(9) of Code Case 1606 NC-3652.2	$2.4\ S_H$	
		Normal & Upset	TH	Eq.(10) of NC-3652.3(a)	S_A	
			P + W + TH	Eq.(11) of NC-3652.3(b)	$S_A + S_H$	
		Vessel Loading Conditions				
		Upset	P + W + OBE	NC-3300 Sec. VIII Code Case 1607	$P_M \le 1.10\ S$ $(P_M \text{ or } P_L) + P_B \le$	1.65 S
		Faulted	P + W +SSE	NC-3300 Sect. VIII, Div. 1	$P_M \le 2.0\ S$ $(P_M \text{ or } P_L) + P_B \le$	2.4 S
	Table 6.1	p. B-5,6				

*Applies to main steam line

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Test and/or analysis Table 6.1, p. 9-1	Not available	Recommended practices for seismic qualification of Class 1E equipment for NPP, IEEE 344, Jan. 1975. Table 6.1 p. 8-1

SEISMIC REVIEW TABLE

Docket Number
50-3

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Indian Point Nuclear Generating Station, Unit No. 1 Reactor type: BWR Containment type: Dry containment-spherical (steel) NSSS Manufacturer: Babcock and Wilcox Architect Engineer: United Engineers and Constructors 5-56/3-62	None	None	Not available	0.10g for containment structure (including steel sphere and interior structure), nuclear service bldg., chemical systems bldg., fuel handling bldg., stack J. Blume p. vi,	.090g for screenwell house 0.03g for superheater bldg. 0.05g for vertical analysis	Synthetic Time History "Earthquake Analysis of Piping Systems." 9-12-69 J. Blume Report, p. 1-2	Each horizontal combined with vertical simultaneously Sheet* 161.1 p. D.2-2	SRSS Sheet 10.1 p. 1-6	Synthetic design spectra TID-7024 Housner "Earthquake and Tornado Analysis of Structures" 9-5-69 J. Blume Report p.1-2	Time-history method J. Blume Report on Piping Systems, p. 1-2 Class I structure Sheet 10.1, p. 1-4, 5 Piping Sheet 11.1, p. 1-2, 5 Reanalyzed, Sheet 4.30, p. 1,2,3

* "Sheet" refers to microfiche Sheet #

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat. Sheet 10.1 p. 2-1	Hard, well-bedded dolomitic limestone, bedrock is extremely jointed and fractured joint systems extended at near right angles to bedding, other systems are irregular. The intensity is almost brecciation.	Foundation sits on bedrock Sheet 10.1 p. 2-1	Not available	Not available	Not available	Stick model with foundation rigidly fixed to bedrock. Sheet 10.1 p. 2-1	Not available	No damping assumed Sheet 10.1 p. 2-1	Not available

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete 5.0/5.0 Structural steel - bolted 2.0/2.5 - welded 1.0/1.0 Sheet 10.1, p. 1-2 Sheet 430, p. 1	First analysis- C = (1.0 ± 0.05) D + (E or W) C = Required load capacity; E = earthquake loads D = Normal loads (dead load of structure, plus any normal operating live loads) Sheet 10.1, p. 1-3 Reanalysis- $U = D + L + F_{eqs} + T_a = P_A$ - steel containment $U = D + L + T_a + F_{eqs}$ - Biological shield $U = D + L + F_{eqs}$ - other Class I structures D = Dead loads; L = live loads T_a = Thermal loads; P_A = pressure loads F_{eqs} = SSE loads Sheet 4.30, p. 1 and 2 Sheet 114.2, Question 7	ACI Standard- ACI 318-63 "Ultimate Strength Design" ASME BPVC, Sec. VIII Sheet 4.1, p. 1-4 Sheet 10.1, p. 1-3 and Sheet 10.2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA	
			LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping	0.5/0.5	Analytical	Reanalysis $U = D + P + P_A' + F_{eqs}$ - piping $U = D + L + F_{eqs}$ - component supports D = Dead doads L = Live loads P = Internal pressure loads P_A' ="Load on safeguard systems in the event of LOCA" F_{eqs} = SSE loads	ASME - USA Standards , code for pressure piping, nuclear power piping, USAS B31.7 also ASME BPVC, Sec. III
Sheet 11.1, p. 1-3 Sheet 430,p. 1		Sheet 5 Sec. 2.1.2.1 p. 4	Sheet 430 p. 1 and 2	Sheet 11.1 p. V. and Sheet 5, Sec. 3.0 p. 8

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-247

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Indian Point Nuclear Generating Station, Unit No. 2 Reactor type: PWR Containment type: Atmospheric (Reinforced Concrete) NSSS Manufacturer: Westinghouse Architect Engineer: United Engineers & Constructors 10-66/10-71	0.10 Sec. 1.2.2 p. 1.2-9	0.05	VI Sec. 2.8 p. 2.8-1	0.15 Sec. 1.2.2 p. 1.2-9	0.10	None used App. A	Horizontal and vertical. acting simultaneously p. A-3	SRSS Q. 1.3-2 Suppl. 9 (5/70)	Housner Fig. A.1-2	No floor response spectra generated; ground response spectra used for piping and components. Sec. 3.1.5 p. 3.0-9 Supp. 6 (2/70)

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Mat foundation 9ft. thick. Sec. 1.3.0 p. 1.0-4 Supp. 6 (2/70)	Hard, wellbedded dolomitic limestone. This bedrock is extremely jointed and fractured. Joint systems extended at near right angles to bedding, other systems are irregular. The intensity may be described almost as brecciation. Sec. 2.7 p. W-4	Not available	Not available.	Stony Point: about 35ft. depth Rockland County 100ft. to 300ft. depth At the fringe of Westchester County depth less than 50ft. Vol. 1, Sec. 2.5, p. 5-10	Not available.	Structure: Stick Model Fixed base Sec. 3.1.5, p. 3.0-9, Suppl. 9	Not available.	Not available	Not available.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>Containment structure 2.0 *

Concrete support structure of reactor vessel 2.0

Steel assemblies:
(a) bolted or riveted 2.5
(b) welded 1.0

Concrete structures above ground
(a) shear wall 5.0
(b) rigid frame 5.0

* One damping value is given, but not clear whether for O.B.E. or D.B.E.</td><td>a) C = 1.0D ± 0.05D + 1.5P + 1.0 (T + TL)
b) C = 1.0D ± 0.05D + 1.25P + 1.0 (T′+ TL′) + 1.25E
c) C = 1.0D ± 0.05D + 1.0P + 1.0 (T" + TL") + 1.0E′
C = Required load capacity section
D = Dead load of structure and equipment loads
P = Accident pressure load
T = Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 x (accident pressure)
TL = Load exerted by the liner based upon temperature associated with 1.5 x (accident pressure)
T′ = Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.25 x (accident pressure)
TL′ = Load exerted by the liner based upon temperature associated with 1.25 x (accident pressure)
E = Load resulting from operational basis earthquake
T" = Load due to maximum temperature gradient through the concrete shell, and mat based upon temperature associated with the accident pressure
TL" = Load exerted by the liner based upon temperature associated with the accident pressure
E′ = Load resulting from design basis earthquake</td><td>ACI 318-63</td></tr>
<tr><td>Sec. 5.1.3.8, p. 5.1.3-6</td><td>Sec. 2.1.12, p. 2.0-5, Supp. 6</td><td>Sec. 2.1.12, p. 2.0-7
and
Sec. 2.1.13, p. 2.0-8
Supp. 6</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE (% critical damping)	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA**				
		LOAD COMBINATION				**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
		L. C.	Vessel	Piping	Supports	
Vital Piping Systems 0.5* * One damping value is given. But not clear whether for O.B.E. or D.B.E.	Analytical and Testing	1. Normal loads	$P_M \le S_M$ $P_L + P_B \le 1.5\ S_M$	$P_M \le S$ $P_L + P_B \le S$	Working stress or applicable factored load value	For mechanical: ASME , BPVC, Section III For piping: USAS B31.1 (1955) For further details refer to Q. 4.10
		2. Normal + Design E.Q.	Same as above	$P_M \le 1.2\ S$ $P_L + P_B \le 1.2\ S$	1 1/3 working stress	
		3. Normal + SSE	$P_M \le 1.2\ S_M$ $P_L + P_B \le 1.2\ (1.5\ S_M)$	$P_M \le 1.2\ S$ $P_L + P_B \le 1.2(1.5\ S)$	Maintain equip. within stress limits	
		4. Normal + pipe rupture	Same as above	Same as above	Same as above	
Sec. 5.1.3.8, p. 5.1.3-7	Sec. 5.1.3.8 p.5.1.3-6 and Q.4.5, Q.4.5-1 Supp. 6	Table A.3-1				Sec. 3.2.3, p. 3.2.3-3 Sec. Q. 4.5, p. Q. 4.5-1 Supp. 6

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-286

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Indian Point Nuclear Generating Station, Unit No. 3 Reactor type: PWR Containment type: Atmospheric (reinforced concrete) NSSS Manufacturer: Westinghouse Architect Engineer: United Engineer and Contractors	.10	.05	VII	.15	.10	Compared with (1) El Centro 12/30/34 and 5/18/40 (2) Olympia 4/13/49 (3) Taft 7/21/52.	3 components: Each horizontal combined with vertical component by absolute sum.	SRSS; closely spaced (10%) modes combined by absolute sum.	Containment response: Housner spectra	Time history.
8-69/5-76	Sec. 5.1.2.2 p. 5.1.2-4	Sec. 5.1.2.2 p. 5.1.2-4		Sec. 5.1.2.2 p. 5.1.2-4	Sec. 5.1.2.2 p. 5.1.2-4	p. A1-9, Appendix A1 Curves-Fig. A1-1&2	Question 5.22	p. Q5.28-1 p. Q5.37-1	Sec. 5.1.3.5 p. 5.1.3-3	p. Q4.32-1 Vol. VI

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Concrete base mat--9 feet thick.	Fine-grained phyllite, a schist, and limestone with bedrock lying close to the surface. Bedrock is jointed and fractured.	Not available.	Not available.	Fluctuates between El. 35 to El. 55 (MSL)	Three reservoirs are within five mile radius. No information on dams is available.	Structure: stick model Soil: cantilever beam assumption indicates fixed base modeling.	Not available.	Not available.	Not available.
Sec. 5.1.2.1 p. 5.1.2-1	Sec. 2.7 p. 2.7-1			See Fig. 2.7-3	Sec. 2.5 p. 2.5-2	Appendix 5A Sec. 3.1.5 p. 5A-26→28			

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Containment: Concrete support structure of reactor vessel: Concrete structures above ground: (a) shear wall (b) rigid frame Steel assemblies: (a) bolted or riveted (b) welded Sec. 2.1.8, p. 5A-10, Appendix 5A Table A.1-1, p. A1-10, Appendix A1	2.0/5.0 2.0/2.0 5.0/5.0 5.0/5.0 2.5/2.5 1.0/1.0	Containment factored load equations: (a) $C=1.0D\pm0.05D+1.5P\pm1.0(T+TL)$ (b) $C=1.0D\pm0.05D+1.25P+1.0(T'+TL')+1.25E$ (c) $C=1.0D\pm0.05D+1.0P+1.0(T''+TL'')+1.0E'$ (d) $C=1.0D\pm0.05D+1.0W'$ (a) = LOCI (b) = Design base accident (DBA)+OBE (c) = DBA+SSE (d) = Design base tornado where C = required load capacity D = dead loads P = accident pressure load T = maximum temperature gradient load associated with 1.5P. TL = liner load due to temperature associated with 1.5P. W' = tornado wind and external pressure drop T' and TL' are T and TL but due to 1.25P. T'' and TL'' are T and TL but due to 1.0P. E = operational base earthquake load E' = design base earthquake load p. 5A-13 Appendix 5-A Table 3.2, 4.1	Containment concrete-- ACI-318-63 Ultimate strength design ACI 318-63 Part IV-B p. 5.1.1-2 p. 5A-13, Appendix 5-A

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="5">MECHANICAL & PIPING</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th rowspan="2">METHOD OF QUALIFICATION</th><th colspan="3">DESIGN CRITERIA</th></tr>
<tr><th colspan="2">LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>Piping: 0.5/0.5

Table A.1-1, p. A.1-10
p. 4.2-8; p. 4.3-29</td><td>Analytical.

Sec. 2
p. A.3-3
p. A.3-10-12
For testing
p. Q4.17
Vol. VI</td><td colspan="2"><table>
<tr><th></th><th>Piping</th><th>Vessels</th></tr>
<tr><td>(1) Normal=D+T+P</td><td>$P \leq \sigma$</td><td>$P_m \leq S_m$ & $P_L \leq 1.5S_m$
$P_m (\text{or } P_L)+P_B \leq 1.5S_m$</td></tr>
<tr><td>(2) Upset=D+T+P+E</td><td>$P \leq 1.2\sigma$</td><td>$P_m (\text{or } P_L)+P_B+Q \leq 3.0S_m$</td></tr>
<tr><td>(3) Faulted=D+T+P+E'</td><td>Design limit curves</td><td>$P_m \leq (1.25S_m)$ or S_y
or
$P_L \leq (1.25S_m)$ or $1.5S_y$
whichever is larger</td></tr>
<tr><td>(4) Faulted=D+T+P+PR</td><td>Design limit curves</td><td>$P_m (\text{or } P_L)+P_B \leq 1.5(1.2S_m)$ or $1.5S_y$ whichever is larger</td></tr>
<tr><td>(5) Faulted=D+T+P+E'+PR</td><td>Design limit curves</td><td>For stress limit refer to Table A.1-3</td></tr>
</table>
D = dead load, T = thermal load, P = pressure load,
E = OBE, E' = SSE

Sec. 4.0, p. A1-18, Appendix A1</td><td>Piping: ANSI B31.1-1955
ASME BPVC
Sec. III-1965

Table 4.10-6 & 10
p. 4.9-2</td></tr>
</table>

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and testing. Sec. 3 Appendix A3 p. Q5.16-2 Vol. VI	Not available.	Westinghouse Report WCAP-7817 "Seismic Testing of Electrical and Control Equipment" Sec. 3 p. A.3-6, 7, 8 Appendix A3 Supplement 4

SEISMIC REVIEW TABLE

Docket Number
50-333

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
James A. Fitzpatrick Nuclear Power Plant Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Stone and Webster Engineering Corp.	0.08	.053	VIII	0.15	0.10	Articifical time-history used	3 components. Results for earthquake in X and Y(vertical) directions simultaneously, and Z and Y directions simultaneously were computed separately.	SRSS	Housner	Time-history method.
5-70/10-74	p. 2.6-1	p. 2.6-1		p. 2.6-1	p.2.6-1	Sec. 2.6 , p. 2.6-1	App. C 3.3 p. C.3-4	Sec. 12.5.1 p. 12.5-1	Sec. 2.6, p. 2.6-2 See Fig. 2.6-1 and Fig. 2.6-2	Sec. 12.5.4, p. 12.5-13

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat. 5'-9" thick embedded 45 ft. below top of bedrock in the surrounding area Sec. 12.3.1, p. 12.3-1	10-12ft. of glacial till which lies directly on top of the Oswego sandstone with laminations and lenticular beds of dark gray shale At 130 ft. below surface it makes contact with Lorraine Group Sec. 2.5 p. 2.5-1	150 ft. of Oswego sandstone	Not available.	Water table at the site slopes toward Lake Ontario at an average gradient of 37 ft. per mile and the direction of ground water is toward the lake. Sec. 2.4.1 p. 2.4-1	Not available.	Stick model with springs to model the rock. Sec. 12.5.1.1 p. 12.5-1	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

STRUCTURES					
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA			
		LOAD COMBINATION			ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
		L. C.	Structural steel	Concrete	
Concrete structures	2.0/5.0	1. Normal dead + live load	AISC Code	ACI 318 working stress	Building code requirements ACI-318 (working stress design)
Steel frame structures,		2. "1" + wind	1/3 increase of AISC	1/3 increase per ACI Code	Specific for structural concrete ACI-301
Bolted and riveted assemblies	2.0/3.0	3. "1" + OBE	Same as above	Same as above	
Welded assemblies	1.0/1.0	4. "1" + DBE	90% of yield	75% of ultimate	Concrete chimneys ACI-307
Fluid containers	0.5/0.5				AISC
		5. Normal dead + tornado load	Same as above	Same as above	NY State Building Construction Code
		6. Normal dead + max. possible flood	Same as above	Same as above	
Sec. 12, Table 12.4-2			Table 12.4.3		Sec. 12.4.8 to 12.4-5

SEISMIC REVIEW TABLE

MECHANICAL & PIPING

	DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA — LOAD COMBINATION	DESIGN CRITERIA — ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital piping systems	0.5/1.0	Analytical	<u>Piping</u>: 1. General membrane primary stress: $S_{LP} + S_{DL} \le S_m$ 2. Operating basis earthquake: $S_{LP} + S_{DL} + S_{OBEQ} = S_{LP} + \frac{M_R}{SM} i \le 1.8\ S_m$ where $M_R = \sqrt{(M_{x1} \pm M_{x2})^2 \pm (M_{y1} \pm M_{y2})^2 + (M_{z1} \pm M_{z2})^2}$ 3. Design basis earthquake $S_{LP} + (S_{DL} + S_{TH} + S_{DBEQ}) = S_{LP} + \frac{M_R}{SM} i \le 3\ S_m$ where $M_R = \sqrt{(M_{x1} + M_{x2} \pm M_{x3})^2 + (M_{y1} + M_{y2} \pm M_{y3})^2 + (M_{z1} + M_{z2} \pm M_{z3})^2}$	<u>For piping</u>: ANSI B31.1.0 App. C.3.3, p. c.3-3 <u>Mechanical</u>: ASME BPVC Section III Subsection B, 1968 Edition and Addenda published to June 30, 1968.
Sec. 12, Table 12.4-2		Sec. 12.5.4, p. 12.5-11	S_{LP} = Longitudinal Pressure Stress S_{DL} = Dead Load Stress S_{TH} = Thermal Stress S_{OBEQ} = Operating Earthquake Stress S_{DBEQ} = Design Earthquake Stress S_m = Allowable Stress at operating temperature i = Appropriate stress intensification factor SM = Section modulus	App. I.3.2.2, p. I.3-2

Section 12.5.4, p. 12.5-10 to p. 12.5-11

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-348

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Joseph M. Farley Nuclear Power Plant Units I and II Reactor type: PWR Containment type: 3 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Bechtel	0.05	0.033	VI	0.10	0.067	Synthesized time history.	3 components: Each horizontal combined with vertical component.	SRSS Closely spaced modes are combined absolutely	Modified Newmark curves.	Time history method.
Unit I: 8-72/6-77 Unit II: 8-72/6-77	Sec. 2.5.2.11	Sec. 2.5.2.11	Sec. 2.5.2.10 p. 2.5-33	Sec. 2.5.2.10 p.2.5-33	Sec. 2.5.2.10 p.2.5-33	Sec. 3.7.1.2 p. 3.7-2	Sec. 3.7.3.7 p. 3.7-14	Sec. 3.7.3.3.4 p. 3.7-13	Sec. 3.7.1.1 p. 3.7-1	Sec. 3.7.2.1 p. 3.7-6

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Rigid mat foundation 9 ft. thick on Lisbon formation. Sec. 2B6.2 p. 2B-15 Sec. 3.8.1.1 p. 3.8-1	Upper residium. Lower residium. Moody's limestone Lisbon formation Sec. 2B.4.3.2 p. 2B-8	40 ft 30 ft 10 ft Sec. 2B.4.3.2 p. 2B-8	Not available.	Approximately 55-65 ft below grade. Sec. 2B.4.3.2 p. 2B-8	There are 13 dams upstream, 14 dams in area: Jim Woodruff, Columbia, Walter F. George, Eagle, City Mills, North Highlands, Oliver, Goat Rock, Bartlett's Ferry, Riverview, Langdale, West Point, Morgan Falls, and Buford Dams. Fig. 2.4-14	Stick model with soil springs. Sec. 3.7.1.6 p. 3.7-3	Soils- 3,000-21,000 psi Lisbon- 50,000-970,000 psi Sec. 2B.7.2.2 p. 2B-20	0.04 critical damping for OBE. 0.07 critical damping for SSE. Table 3.7-1	Not available.

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Welded steel frame structures:	2.0/5.0	Design loading case:	ACI 318-63
Reinforced concrete structures plus equipment supports:	2.0/5.0	1. D+F+L (construction case)	AISC 1969
Prestressed concrete structures:	2.0/5.0	2. $D+F+L+T_o+E$ (or W) (operating case)	AEC Reg. Guides
		3. $D+F+L+P+T_e$ (design accident case)	For further details refer to Section 3.8.1.2.
		4. $D+F+L+T_s+E$ (or W) (prolonged shutdown case)	
		5. D+F+L+1.15P (test case)	
		Factored loading case:	
		1. $C=1/\phi(1.0D+1.5P+1.0T_a+1.0F)$	
		2. $C=1/\phi(1.0D+1.25P+1.0T_a+1.25H+1.25E$ (or 1.25W) $+1.0F)$	
		3. $C=1/\phi(1.0D+1.25H+1.0R+1.0F+1.25E$ (or 1.25W) $+1.0T_o)$	
		4. $C=1/\phi(1.0D+1.25H+1.0F+1.25W_t+1.0T_o)$	
		5. $C=1/\phi(1.0D+1.0P+1.0T_a+1.0H+1.0E'+1.0F)$	
		6. $C=1/\phi(1.0D+1.0H+1.0R+1.0E'+1.0F+1.0T_o)$	
Table 3.7-1		Sec. 3.8.1.3 p. 3.8-13	Sec. 3.8.1.2 p. 3.8-3

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION		DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
			L. C. -Class 1 Components	Stress Limits	ASME, BPVC, Section III, Table 3.9-3
Vital piping:	0.5/1.0	Analytical and Testing	Normal	$P_M \leq S_M$ $P_L \leq 1.5\ S_M$ P_M (or P_L) + $P_B \leq 1.5\ S_M$ P_M (or P_L) + P_B + Q $\leq 3.0\ S_M$	"Design Criteria for Components not covered by ASME Code." Ex. Heat exchangers - ARI 410-64 Fan AMCA Test Code 300-67, 211 A-67
Welded steel plate assemblies:	1.0/2.0				
Bolted and riveted steel:	3.0/5.0		Upset	Same as normal	
			Faulted	Table 5.2-6	
Table 3.7-1		Sec. 3.7.2.1 p. 3.7-5 3.9-1, 3.9-24 3.9-3	p. 3.9-1, Table 3.9-1, Table 5.2-4, -5, -6, -7		Table 3-9-3 Section 3.9.2, 3.9.2

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Testing and analysis. Sec. 3.10.1 p. 3.10-2	For electrical cable tunnels: (Dead load + live load + E.Q.) 0.75 $\leq$ maximum allowable stress Table 3.8-14	IEEE 344-1971 Sec. 3.10.1,2 p. 3.10-2,3

SEISMIC REVIEW TABLE

Docket Number
50-305

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Kewaunee Nuclear Power Plant Reactor type: PWR Containment type: Dry containment-cylindrical (steel) NSSS Manufacturer: Westinghouse Architect Engineer: Pioneer 8-68/12-73	0.06 App. B Sec. B.4.5 p. B.4-2	0.04 App. B Sec. B.6.3 p. B.6-5	V normal focus shock within 7 miles of plant site. VII normal focus shock App. A p. 31-32	0.12 App. B Sec. B.4.5 p. B.4-3	0.08 App. B Sec. B.6.3 p.B.6-6	Synthetic time history	Horizontal and vertical components Combination not known App. B p. B.6-5	SRSS App. B p. B.6-5	Newmark method Plate 8-A and Plate 8-B App. A p. 33	Spectral method Blume report #JAB-PS-01, JAB-PS-03 App. B p. B.6-5

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Soil-bearing type (Raft-type formation) Concrete base slab 35 ft. depth of slab	Glacial till Glacial lacus-trine deposits Bedrock (Niagra dolomite)	60-150 ft 350-600 ft	Shear wave velocity soil =2500 fps Shear wave velocity rock =11,500 fps	Varies from 10-30 ft below ground surface	Not avail-able.	Stick model with soil springs.	Glacial till $G=1x10^7$ lbs/sq ft Glacial lucustrine deposits $G=5x10^5$ lbs/sq ft Bedrock $G=7.5x10^8$ lbs/sq ft	5% critical damping OBE,SSE	Not avail-able.
App. E Sec. E.1-E.3 Fig. E.2-5	App. A p. 16	App. A p. 16	App. A p. 16	App. A p. 11		App. B Sec. B.6.3 p. B.6-5	App. A p. 26 - Table 7	App. B Table B.6-5	

SEISMIC REVIEW TABLE

STRUCTURES				
		DESIGN CRITERIA		
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reactor Containment vessel	1.0/1.0			ACI 318-63
Shield building	2.0/2.0	Normal operating	Dead+live+wind+snow	
Reactor containment vessel internal concrete	5.0/5.0	OBE	Dead+live+DBA+snow+greater of the OBE or wind	
Steel frame structures	2.0/2.0	DBE	Dead+live+snow+DBA+DBE	
Reinforced concrete construction	2.0/2.0	Tornado	Dead+live+300 mph design tornado+tornado missile, if any	
App. B Table B.6-5		Table B.6-1		App. B Table B.6-2

MECHANICAL & PIPING						
DAMPING OBE/SSE	(% critical damping)	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA** — **LOAD COMBINATION**			**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
				Pressure Vessels	Piping	
Piping systems	0.5/0.5	Analytical or Tests.	Normal condition:	(a) $P_m \leq S_m$ (b) P_m(or P_L)+$P_b \leq 1.5S_m$ (c) P_m(or P_L)+P_b+$Q \leq 3.0S_m$	$P \leq S$	ASME, BPVC, Sec. III, 1968 ANSI B31.1 code for power piping 1967.
Mechanical Equipment	2.0/2.0		Upset condition:	(a) $P_m \leq S_m$ (b) P_m(or P_L)+$P_b \leq 1.5S_m$ (c) P_m(or P_L)+P_b+$Q \leq 3.0S_m$	$P \leq 1.2S$	
			Emergency condition:	(a) $P \leq 1.2S_m$ or S_y (b) P_m(or P_L)+$P_b \leq 1.8S_m$ or $1.5S_y$	$P \leq 1.5(1.2S)$	
			Faulted condition:	(a) Stainless steel: design limit curve (b) Carbon steel: (i) P_m=$1.5S_m$ or $1.2S_y$ (ii) P_m(or P_L)+$P_b \leq 2.25S_m$ or $1.875S_y$	(a) Stainless steel design limit curve (b) Carbon steel $P \leq S_y$ or 1.8S	
App. B Table B.6-5		App. B p. B.7-10d,e	Table B.7-2 Table B.7-3 For further details refer to App. B			App. B p. B.7-6

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Analysis	"Electrical equipment and its supports were designed to be sufficiently rigid so that its natural frequency will be out of the range of resonance with the building structure".	Not available

B.7-10C

SEISMIC REVIEW TABLE*

Docket Number
50-409

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
La Crosse (Genoa) Nuclear Generating Station Reactor type: BWR Containment type: Pre-Mark (steel) NSSS Manufacturer: Allis Chalmers, Manufacturing Co. Architect Engineer: Sargent and Lundy Engineers 3-63/7-67	.06 Sec. 2.4	.04 (Vertical acceleration used for re-analysis of M.S., Feedwater HPCI piping systems 1975-77). Sec. 2.4	VI Sec. 2.4	.12 Sec. 2.4	.08 Sec. 2.4	Taft 1952 record chosen as initial accelerogram. A ground time-history which envelops the 2% damping curve of R.G. 1.60 was generated for analysis of major structures such as the containment.	Horizontal only for RCB Maximum horizontal spectra (x or z direction) are added simultaneously with the vertical for major piping and equipment.	SRSS for equipment and piping (R.S.) Algebraic sum for reactor bldg. (time history method.)	R.G. 1.60 used as basis to develop response spectra from Taft earthquake. (not specifically stated as such but curves are those of R.G. 1.60)	No vertical response spectra generated, instead use 2/3 of horizontal ground response spectra. Horizontal response spectra derived from time history analysis. Reanalysis of Mechanical and Piping, 1975-77, No amplification of vertical response.

*Information was obtained from BNL Docket search and SEPB Report "Seismic Review of La Crosse BWR Phase I Report."

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Pile foundation 232 piles will support 50 tons each	15 ft. of hydraulic fill overlies about 100-130 ft. of glacial outwash and fluvial deposits at the site. Bedrock of flat-lying sandstone and shale of the	Dresbach group extends below these deposits about 650 ft. where it makes contact with the crystalline basement.	Not available	Not available	Not available	Lumped-mass for structure soil-spring and dashpot deconvolution process used; soil layers modeled as shear beam (2% damping used)	Not available	Not available	Not available

SEISMIC REVIEW TABLE

STRUCTURES

	DAMPING OBE/SSE 1/2 SSE	(% Critical damping) SSE
Reactor Containment	3.0 up	7.0 up
Turbine building		7.0
Stacks		7.0 up
New diesel generator building	4.0	7.0

DESIGN CRITERIA

LOAD COMBINATION

Structural Steel - Elastic:

Construction: $1.0\ D + 1.0\ L + 1.0\ T_o + W < 1.33$ AISC (1969)

Test: $1.0\ D + 1.0\ L + 1.0\ T_o + 1.0\ R_o < 1.33$ AISC (1969)

Normal: $1.0\ D + 1.0\ L + 1.0\ T_o + 1.0\ R_o <$ AISC

Severe Environmental: $1.0\ D + 1.0\ L + 1.0\ T_o + 1.0\ R_o + E <$ AISC

Extreme Environmental: $1.0\ D + 1.0\ L + 1.0\ T_o + 1.0\ R_o + E' < 1.6$ AISC

R/C - strength design:

Construction: $1.1\ D + 1.3\ L + 1.3\ T_o + 1.3\ W$

Test: $1.1\ D + 1.3\ L + 1.3\ T_o + 1.3\ R_o$

Normal: $1.4\ D + 1.7\ L + 1.3\ T_o + 1.3\ R_o$

Severe Environmental:
$1.4\ D + 1.7\ L + 1.3\ T_o + 1.3\ R_o + 1.3\ W$
$0.9\ D + 1.3\ T_o + 1.3\ R_o + 1.3\ W$
$1.4\ D + 1.7\ L + 1.3\ T_o + 1.3\ R_o + 1.4\ E$
$0.9\ D + 1.3\ T_o + 1.3\ R_o + 1.4\ E$

Extreme Environmental: $1.0D + 1.0\ L + 1.0\ T_o + 1.0\ R_o + 1.0\ E'$

Section 3.7.1; Table 4.5-1 and 4.5-2

ACCEPTANCE CRITERIA & ALLOWABLE STRESSES

Allowable structural Capacities for RCB, Two stacks, turbine building waste disposal building:

Concrete:	1/2 SSE	SSE
Moment	M_u	$0.63\ M_u$
Shear	V_u	$0.60\ V_u$
Steel		
Moment	$0.66\ M_y$	M_y
Shear	$0.40\ V$	$0.53\ V$

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
	DAMPING OBE/SSE (% Critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping	1/2 SSE 1.0	SSE 2.0	Not available	M.S. Piping: Load conditions from NB-3110, 3620 Design: (Primary) $P_o + DL + E < 1.5\ S_M$ Normal: (Primary and secondary) $T + P + SA + TA + E < 3\ S_M$ Upset: Same as for normal condition Emergency: (Primary stress) $< 2.25\ S_M$ Faulted: $P_o + DL + E < 3.0\ S_M$ (Main steam piping and feedwater piping designed as Class 2 since fatigue loads not considered). Follows R.G. 1.48, EQ 8,9,10,11 of ASME Code	Piping: AEC Reg. Position 1 and Subsection NB-3600 of Section III of ASME B&PV Code

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-309

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Maine Yankee Atomic Power Company Reactor type: PWR Containment type: Sub-atmospheric (Reinforced concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Stone & Webster Engineering Corp.	0.05	0.033	VI	0.10	.067	No earthquake time-history used.	Each horizontal combined with the vertical resulting in two load cases. The method of combination is unclear.	No combination used flexual mode used only.	Housner spectra	Empirical procedure used for piping to provide amplified response spectra. For equipment and anchors used equivalent static load method or Housner response spectra. Amendment 22 (4-71) Q. 4.4 Q. 4.5 Method used described in Section 5.1.1.2.2 p. 5-6
10-68/9-72	Sec. 1.3.2 p. 1-6	Sec. 1.3.2 p. 1-6		Sec. 1.3.2 p. 1-6	p. 1-6	Amendment 20 (3-71) Q. 4.5	p.5-3	p. 5-6	Sec. 2.5.4 p. 2-27 Figs. 2.5.6 and 2.5.7	

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Flat reinforced concrete slab bearing on bedrock with a central reactor vessel pit. 10 ft. thick Sec. 5.1 p. 5-1	Major structure Hard crystalline bedrock Minor structure on rock or compacted granular fill above the rock. Sec. 2.4 p. 2-23	Joints in bedrock are medium spaced, ranging from 1 to 5 ft intervals or and less. Sec. 2.4 p. 2-23	7,000 fps Sec. 2.4 p. 2-23	Dug wells: less than 25 ft deep. Drilled wells: depth of 100 ft or more. Sec. 2.3.3 p. 2-22	Not available.	Translational & Rocking modes were not incorporated in the dynamic model. Sec. 5.1.1.2.2 p. 5-6	$1.80 \times 10^6 - 2.06 \times 10^6$ psi Sec. 2.4 p. 2-23	Not available.	Not available.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr>
<td>1. Reactor containment. 5.0/7.0

2. Reinforced concrete structure, other than containment (on rock or soil). 5.0/7.0

3. Reinforced concrete structure (not on soil or rock). 2.0/5.0

4. Steel framed structure
Bolted or riveted 3.0/5.0
Welded 1.0/2.0

5. Reactor vessel
Welded assemblies 1.0/1.0
Bolted assemblies 3.0/3.0

Table 2.5-1</td>
<td>1. (1.0 ± 0.05) D + 1.5 P + 1.0 (T+TL)
2. (1.0 ± 0.05) D + 1.25 P + 1.0 (T+TL) + 1.25 E
3. (1.0 ± 0.05) D + 1.0 T + 1.0 C
4. (1.0 ± 0.05) D + 1.0 P + 1.0 (T+TL) + 1.0 E'

D = dead load
P = design pressure load
TL = load by exposed liner
T = temperature gradient load
E = OBE
E' = SSE

Section 5.1.1.2, p. 5-2</td>
<td>Containment:
Ultimate strength methods ACI 318-63, Sec. 1504, Part IV B or the Ultimate Strength Design Handbook ACI Special Publication No. 17.

Section 5.1.1.2, p. 5-2</td>
</tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA LOAD COMBINATION *	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Mechanical equipment. 2. Piping.	2.0/2.0 1.0/2.0	Analytical	Reactor vessel internal structure 1. Design loading + OBE — $P_m \leq S_m$, $P_B + P_L \leq 1.5\ S_m$ 2. Normal Operating + SSE — $P_m \leq S_D$, $P_B \leq 1.5\ [1-(\frac{P_m}{S_D})^2]S_D$ 3. Normal Operating + SSE + pipe rupture — $P_m \leq S_L$, $P_m \leq 1.5\ [1-(\frac{P_m}{S_L})^2]S_L$ Where: $S_L = S_y + (1/3)(S_u - S_y)$ $S_D = 1.2\ S_m$	ASME BPVC, Section III
			Piping Applicable code allowables 1. Design load + OBE 2. N.O. + SSE — $P_m \leq S_D$, $P_B \leq \frac{4}{\pi} S_D \cos(\frac{\pi}{2} \cdot \frac{P_m}{S_D})$ 3. N.O. + SSE + pipe rupture — $P_m \leq S_L$, $P_B \leq \frac{4}{\pi} S_L \cos(\frac{\pi}{2} \cdot \frac{P_m}{S_L})$	
Amendment 20 (3-71) Q. 4.9, Table 2.5-1		Amendment 22 (4-71) Q. 4.8		p. 3-4, 4.2-4

*For reactor internals: Table 3.2-1, p. 3-4
Vessels and piping: Table 4.2-3, p. 4.2-4

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE *

Docket Number
50-245

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Millstone Point Nuclear Power Station Unit 1 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Ebasco 5-66/10-70	0.07 Sec. XII p. XII-1.7	0.05 Sec. XII p. XII-1.7	VII	0.17 Sec. XII p. XII-1.7	0.113 p. XII-1.7	Taft 69° west earthquake record (Blume response spectrum is more conservative than Taft response spectrum) Q VII - A.9 and Q VII - A.10 Amend. 17	Horizontal and vertical (X+Y,Z+Y) The resulting seismic stress for the two motions were combined linearly. Sec. XII p. XII-1.7	No modal combination needed for time history. Unclear information for response spectrum method.	Housner Fig. XII-1.2 Fig. XII-1.3 Sec. XII p. XII-1.7	Equivalent Static Method - for intake structure, turbine bldg., main steam lines, Class I piping in reactor and turbine bldg., batteries and battery racks. Time History Method: Reactor bldg., ventilation stack, radwaste/control room, condensate storage tank Response Spectrum Gas turbine bldg., recirculation loop piping, torus, RPV, isolation condensor, fuel racks p. XII-1.12

Information obtained from BNL Docket Search and SEPB Report, "Seismic Review of Millstone Nuclear Power Station, Unit 1"

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete square mat (42'-6") and six feet of thickness at elevation of 32'-0". The foundation is supported directly on the bedrock. Gas turbine building founded on piles. Turbine build mat foundation on piles.	not applicable	not applicable	14,000 fps Sec. XII-p. XII-1.13	Not available	None	Lumped mass with soil springs (for reactor bldg. only). Rocking mode was considered for reactor bldg. Fixed base without rocking for other major structures. Sec. XII p. XII-1.2.1	Not available	Not available	Not available

SEISMIC REVIEW TABLE

STRUCTURES				
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA		
		LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Reinforced concrete structures	5.0	1. D + R + E	- Normal allowable code stresses are used in AISC and ACI increase in design stress for earthquake loads is not permitted.	1. AISC 2. ACI Code
2. Steel frame structures	2.0			
3. Welded assemblies	1.0	2. D + R + E′	- Stresses are limited to the minimum yield point. In few cases, stresses may exceed yield pt. then in this case the limit-design method as discussed in AEC publication TID -7024 "Nuclear Reactor and Earthquakes", Section 5.7, to determine that the energy absorption capacity exceeds the energy input.	
4. Bolded and riveted assemblies	2.0			
5. Ventilation stack	5.0			
6. Radwaste Bldg., Control room	5.0			
7. Condensate storage tank	0.5(fluid) 2.0(tank)			
8. Gas Turbine Bldg.	5.0	D = Dead load	R = Jet force or pressure due to rupture of any one pipe	
		E = Design earthquake load	E′ = maximum earthquake load	
		Sec. XII - 1.12 1. DL + LL + OL + E (.07g) 2. DL + LL + OL + W 3. DL + LL + OL + E′(.17g)		
Sec. XII p. XII-1.7 and Table VII - A.14-1, Q.A.14, Amend. 17		Table XII -1 p. XIII - 1.3		Table XII-1

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="4">MECHANICAL & PIPING</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th rowspan="2">METHOD OF QUALIFICATION</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>1. Vital Piping System 0.5
Sec. XII
p. XII-1.7

2. Containment heat exchange 2.0
3. RPV 2.0
4. Recirculation loop piping 0.5
5. Suppression chamber 2.0</td><td>Analytical</td><td><u>Reactor Vessel Internals</u>
1. D + E Stress criteria of ASME Section III, Class A vessel
2. D + E′ The secondary and primary plus secondary stresses are examined on a rational basis taking into account elastic and plastic strains.
<u>Emergency Core Cooling Systems</u>
1. D + T + H + E Stresses remain within code allowable. USAB-B 31.1 plus code cases (piping)
2. D + T + H + E′ Primary stresses are within the stress criteria of ASME Section III, Class A. The secondary and primary plus secondary stresses and examined on a rational basis taking into account elastic and plastic strains. These strains are limited to preclude failure by deformation.
<u>Primary Containment</u>
1. D + P + H + T + E
2. D + P + R + H + T + E
3. D + P + R + H + T + E
D = Dead load
P = Pressure due to LOCA
R = Jet-force or pressure on structure due to rupture of any one pipe
H = Force on structure due to thermal expansions of pipes
T = Thermal loads on containment due to LOCA
E = Design E.Q. load; E′= maximum E.Q. load</td><td>ASME Section III, Class B
USAS - B31.1-1967

Sec. XII
Question A.14, Amend 17
Table XII-1</td></tr>
</table>

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE TRESSES
Not available	Not available	Battery racks and batteries were designed to withstand lateral and vertical seismic loads of 0.12g horizontal and 0.046g vertical	Not available

SEISMIC REVIEW TABLE

Docket Number
50-336

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Millstone Nuclear Power Plant Unit 2 Reactor type: PWR Containment type: 3 buttresses with shallow dome (pre-stressed concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Bechtel	0.09	0.06	VII	0.17	0.11	Synthetic time-history	3 components: Each horizontal combined with vertical component simultaneously.	Absolute sum method.	Separate sets of design spectra were developed for rock foundation and backfill. Housner for rock foundation. Modified Newmark for backfill	Time history method.
12-70/9-75	Sec. 5.8.1.1 p. 5.8-1	Sec. 5.8.3.2.2 p. 5.8-8	Amend. 39 Sec. 2.6	Sec. 5.8.1.1 p. 5.8-1	Sec. 5.8.3.2 p.5.8-8	Sec. 5.8.1.1 p. 5.8-1 Fig. 5.8-6	Sec. 5.8.4 p. 5.8-11	Sec. 5.8.3.2.11 p. 5.8-7	Sec. 5.8.1 p. 5.8-1 Fig. 5.8-1,2 Fig. 5.8-3,4	Sec. 5.8.4 p. 5.8-11

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reactor building mat rests on unweathered rock. Depth: 8½ feet Sec. 2.7.5 p. 2.7-3 Sec. 5.2.1 p. 5.2-1	Glacial deposits: Ablation till and a dense basal till which lies above the bed-rock. Bedrock consist of Monson gneiss intruded by westerly granite. Sec. 2.4 p. 2.4-4 p. 2.4-5	Glacial deposits 0 to 30 ft Bedrock: 11 to 54 ft below ground. Sec. 2.4 p. 2.4-4 p. 2.4-5	5500-7500 fps in bedrock. Sec. 2.4.4 p. 2.4-9	Little or no ground water is present in bed-rock. So virtu-ally all ground water is restrict-ed to the soil overburden. Water level is subjected to considerable seasonal fluctua-tions. Sec. 2.5.2 p. 2.5-2 Fig. 2.4-2c, 2d	Not avail-able.	Backfill: Stick model with soil springs. Bedrock: Stick model with fixed base. Sec. 5.8.2 p. 5.8-3,4	Not available.	2%/5% Table 5.8-1 p. 5.8-9	2% Sec. 5.8.3.3 p. 5.8-10

SEISMIC REVIEW TABLE

STRUCTURES				
		DESIGN CRITERIA		
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Welded steel plate assemblies:	1.0/1.0	a. D+F+L	Construction case	ACI-318-63
Welded steel framed structures:	2.0/2.0	b. $D+F+L+T_o+E$	Operating case	ACI-301-66
Bolted or riveted steel framed structures:	2.5/2.5	c. $D+F+L+P+T_i$	Design incident case	ASME, BPVC (1968)
Reinforced concrete equipment supports:	2.0/3.0	d. $D+F+L+T_s+E$	Prolonged shutdown case	AISC, 1963
Reinforced concrete frames and buildings:	3.0/5.0	e. D+F+L+1.15P	Test case	
Prestressed concrete structures:	2.0/5.0	D = dead loads		
		L = live loads		
		F = prestressing loads		
		P = design pressure		
		T_i = thermal loads due to the loss of coolant incident		
		T_o = thermal loads due to operating temperature		
		T_s = thermal loads due to transient wall temperature over a prolonged shutdown (20 F at exterior face, 70 F at center, 50 F at interior face)		
		E = operating basis earthquake loads (0.09 g)		
		For further details refer to Section 5.2.3.2.5.		
Table 5.8-1, p. 5.8-9		Sec. 5.2.3.2.4 p. 5.2.8		Sec. 5.1.2 p. 5.1-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
		DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Steel piping: 0.5/0.5	Analytical and testing.	Reactor coolant system (vessels): 1. Design loading + OBE $P_m \leq S_m$ $P_b + P_L \leq 1.5 S_m$ 2. Normal operation + SSE $P_m \leq S_m$ $P_b \leq 1.5 \left[1 - \left(\frac{P_m}{S_D}\right)^2\right] S_D$ 3. Normal operation + SSE + pipe rupture $P_m \leq S_m$ $P_b \leq 1.5 \left[1 - \left(\frac{P_m}{S_L}\right)^2\right] S_L$ $S_L = S_y + (1/3)\ (S_u - S_y)$ R.C.S. (Piping) 1. Design loading + OBE $P_m \leq S_m$ $P_b + P_L \leq 1.5 S_m$ 2. Normal operation + SSE $P_m \leq S_m$ $P_b \leq 4/\pi S_D \ \mathrm{Cos}\left(\frac{\pi}{2} \cdot \frac{P_m}{S_D}\right)$ 3. Normal operation + SSE + pipe rupture $P_m \leq S_L$ $P_b \leq 4/\pi S_L \ \mathrm{Cos}\left(\frac{\pi}{2} \cdot \frac{P_m}{S_D}\right)$	Piping-- ANSI B 31.7 ANSI B 31.1.0 Sec. 1.2.14, p. 1.2-21 and Sec. 4.5.2.1, p. 4.5-5 Pressure vessels-- ASME, BPVC, p. 1.2-19 and Sec. 4.5.2.2, p. 4.5-5
Sec. 5.8.3.3 p. 5.8-9	Sec. 5.8.5 p. 5.8-12	See Table 4.2-2, p. 4.2-3. For mechanical see Sec. 3.2.1, p. 3.2-1 to 3.2-5.	

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and testing. Sec. 5.8.6 p. 5.8-13	Not available.	Instrumentation designed as per Reg. guide 1.12. Sec. 5.8.6 p. 5.8-13

SEISMIC REVIEW TABLE

Docket Number
50-263

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Monticello Nuclear Generating Plant, Unit 1 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel	Class I 0.06 Class II 0.05	0.004 0.0033	VIII	0.12	0.08	Taft Earthquake of July 21, 1952, North 69 West component	Horizontal and vertical component combined linearly.	SRSS	Response spectra from Taft earthquake	Time-history analysis for Class 1 structures. UBC for Class 2
6-67/9-70	Sec. 2.1.9 p. 12-28	Sec. 6.0 p. 2.6-1	Sec. 6.0 p. 2.6-1	Sec. 2.1.9 p. 12-28		Sec. 6.0, p. 2-6.1	Sec. 2.1.9, p. 12-2.8	Sec. 2.1.9 p. 12-2.9c and Vol.VI Append. A Reactor Building Seismic Analysis p-6	Fig. 2-6-5 p. 2-6.1 Sec. 2.1.9, p. 12-2.8a and p. 12-2.9	Sec. 2.1.9 p. 12-2.9

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION: TYPE *	BEARING INFORMATION: THICKNESS	BEARING INFORMATION: V_s PROFILE	GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
Reinforced concrete mat founded on medium sand with some gravel.	Decomposed granitic and basic rocks of precambrian age comprise	the formation. 75 to 122 ft.	Not available.	The water table beneath the low terraces which border the Mississippi River usually lies at about river elevation and slopes very slightly toward the river during periods of normal stream flow.	Not available.	Stick model with soil springs.	Not available.	Not available	10.0% of critical damping.
	Wethered precambrian crystalline rocks are directly below	the above strata. 10 to 15 ft.		Groundwater at shallow depths moves toward the Mississippi River or its tributaries at variable gradients depending on local conditions.					
Sec. 2.2.1.1 p. 12-2.13	Above the sandstone is a series of alluvial strata consists predominately of clean	sands with gravel, as well as a few layers of clay and gracial till. 50 ft.		Sec. 5.4, p. 2-5.3 and Fig. 2-5-3		Append. A. at Seismic Analysis Part p. 2		App. A Sec. 2.1.9 p. 12-2.8	Append. A Table 1 p.8

Sec. 5.3, p. 1-5.2, p. 2-5.3

*Because of space Type and Thickness columns are combined together.

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Recommended damping: Rector-boilding (massive construction with many cross walls and equipment and providing only secondary containment) Thin-shell and prestressed concrete Steel structures Ref. Append. A., Table 1, p.8	 5.0 2.0 2.0	1. Primary containment a. D + P + H + T + OBE b. D + P + R + H + T + OBE c. D + P + R + H + T + SSE 2. Reactor building and all other Class 1 structure a. D + R + OBE b. D + R + SSE c. D + W d. D + W' Sec. 2.1.4, p. 12-2.3 and 12-3.6	AISC - Sixth Edition ACI - 318-63 ASME CODE Sec. III and IX ACI 505-54 for R. C. Chimney Sec. 2.1.4, NSP-1, p. 12-2.6 Table 12-2-1 Sec. 2-1.4, p. 12-2.4 and p. 12-2.5

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping: Vital Damping System 0.5	Analytical	3. Reactor vessel supports a. D + H + R + OBE b. D + H + R + SSE 4. Reactor vessel internals a. D + O.B.E. b. D + S.S.E. c. D + P 5. Emergency core cooling system (ECCS) a. D + O.B.E. b. D + S.S.E. For piping: Suction header pipe: Dead loads + seismic loads + OBE = 820 psi Dead loads + seismic loads + SSE = 1640 psi allowable stress is 17, 500 psi	ASME Sec. III and USAS B 31.1-1967
Append. A, Table 1, p. 8	Sec. 2.1.9, p. 12-28	Sec. 2.1.4, p. 12-2.3-12.2.6 p. 12-2.11	Sec. 2.1.4., p. 12-2.5 and p. 12-2.6

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Inspection and testing for: 1. Auxiliary Power System 2. Plant standby generator systems. 3. D-L Power supply systems. 4. Reactor protection system power supplies. Sec. 8 p. 8.3-5 p. 8.4-4 p. 8.5-6 p. 8.6-2	Not available.	For diesel-generator set: Equipment shall conform to applicable standards of the NEMA, ASA, DEMA, ASME, NBFW, NIPA, ASTM, IEE, USASI and state and local regulations. Sec. 4.1 p. 8-4.1

SEISMIC REVIEW TABLE

Docket Number
50-220

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Nine Mile Point Nuclear Station Unit No. 1 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Stone & Webster Engineering Corp.	Not used	Not used	IX	0.11	0.055	Not used	Not available.	SRSS	Hounser	Analysis by Reserve Energy-Technique, by John Blume
4-65/8-69			PHSR III-1	PHSR III-1	Amendment 6, Supp. 2, Question I-11			Amend. 6, Supp.2, Question I-2.	PHSR VI-22 App. C	PHSR III-1

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
All major structures founded on Oswego sandstone. Reactor bldg. is founded in rock to a depth of 60 ft.	10-12 ft. of glacial till was removed. Bedrock is Oswego sandstone. It makes contact with Lorraine Shale at a depth of 185 ft.	185 ft.	14,000 fps	195 ft. below ground surface	Not available.	Stick model with soil springs.	Not available.	2 to 3% critical damping.	Not available.
PHSR III-3	Amend. 2, Vol. 2, FSAR 6/1/67		Amend. 6, Supp. 2, FSAR, Oct. 1968, Question IV 12, p IV-24	App. C "Earth Science"		Amend. 6, Supp.2, Question I-2		Amend. 6, Supp. 2, FSAR Oct. 1968, Question IV 12, p IV-25	

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
critical damping for integral reinforced-concrete structures...................... 5.0 critical damping for ventilation stack... 7.5 Details: First supplement to PHSR in answer to question III-1(d)	Reactor bldg. Waste disposal bldg. screen and pump house drywell radial steel framing: DL + LL + OL + Design Earthquake Reactor vessel concrete pedestal DL + Equipment Load + Temp. (operating) DL + Equipment Load + Jet Load + Temp. + Design Earthquake See Table I-4 for 10 load combinations for the drywell	1. ACI-318-63 2. For proportioning of concrete members: Part IV-A "Working stress design" of Code 318-63. 3. Reinforced-concrete ventilation stack: ACI 505-54 4. AISC specifications for the design, fabrication and erection of structural steel for building. 5. New York State Building Code 6. UBC
Amendment 6, Supp. 2, Question I-5	Supplement 2, question I-4, question I-9	Amend. 6, Supp. 2, Question I-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Core spray piping and sparger ring located in the reactor vessel: Equations given in ASME Section III. Drywell - ASME Sect. VIII plus Code Case 1270N-5, 1271N, 1272N-5 Amend. 5-Supp 1 (5/20/68) Question II-12.	1. "Method of Differences" 2. Reactor internals: ASME Code Class A 1. Amend. 6, Supp. 2, Question I-10 2. Amend. 5, Supp. 1 FSAR Question I-5

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-338

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
North Anna Power Station Unit 1 Reactor type: PWR Containment type: Sub-atmospheric (reinforced concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Stone and Webster 2-71/ 11-77	0.06g for structures on rock 0.09g for structures on soil p. 1.2-2 1.2-3	0.04g for structures on rock 0.06g for structures on soil p. 1.2-2 p. 1.2-3	VII	0.12g for structures on rock 0.18g for structures on soil p. 1.2-2 1.2-3	0.08g for structures on rock 0.12g for structures on soil p.1.2-2 1.2-3	E-W and N-S components of Helena, Montana 1935 earthquake, and the S-E component of the San Francisco 1957 earthquake. p. 2.5-9	2 components: Horizontal plus vertical added simultaneously p. 3.7-10	SRSS Sec. 3.7	Developed from Helena 1935 and San Francisco 1957 by enveloping the response spectra shown in Fig. 2.5-9 thru Fig. 2.5-12. p. 2.5-9	Time history method. Sec. 3.7

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Flat reinforced concrete mat 10 ft. thick. Founded on concrete backfill. p. 1.2-2 p. 2.5-17	Saprolite soil weathered rock p. 2.5-12	Not available.	Not available.	Not available.	North Anna Reservoir Sec. 2.4.1.1	Stick model with soil springs. Sec. 3.7 p. 2.5-9	Fresh and slightly weathered rock $G=1.0x10^6$ psi Soils @ 10 ft. depth 14,000 psi @ 20 ft. depth 19,800 psi p. 2.5-24	Not available.	Not available.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="5">STRUCTURES</th></tr>
<tr><th colspan="3" rowspan="2">DAMPING
OBE/SSE</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION
Containment Structural Loading Criteria:</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td>Stress Level</td><td>Type & Condition of Struct, Syst. or Component</td><td>Percentage Critical Damping</td><td rowspan="4">(1.0 ± 0.05) D ± 1.0 P ± 1.0 (T + TL) + 1.5 E
(1.0 ± 0.05) D ± 1.0 P ± 1.0 (T + TL) + 1.0 (DBE)
(1.0 ± 0.05) D + 1.25 P + (T' +TL') + 1.25 E</td><td rowspan="4">AISC Manual
ACI 301-66
ACI 318-63</td></tr>
<tr><td>1. Low Stress, well below proportional limit. Stresses below 0.25 yield point.</td><td>a. Steel, reinforced concrete; no cracking and no slipping at joints.</td><td>0.5 to 1.0</td></tr>
<tr><td>2. Working stress limited to 0.5 yield point stress</td><td>a. Welded steel, well reinforced concrete (with only slight cracking)
b. Bolted steel</td><td>2.0

5.0</td></tr>
<tr><td>3. At or just below yield point</td><td>a. Welded steel
b. Reinforced concrete
c. Bolted steel</td><td>5.0
5.0
7.0</td></tr>
<tr><td colspan="3">Table 3.7.2-1</td><td>p. 3.8-87, Table 3.8.2.2-1</td><td>p. 3.7-49
3.8-17</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA	
			LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Piping p. 3.7-23	0.5/1.0	Analysis and Testing p. 3.7-46,47 p. 3.7-22	ASME Class 1 Piping: based on Subarticle NB-3650 Class A Components 1) Normal a) $P_m \le S_m$, b) P $\le 1.5\ S_m$, c) P_m(or P_L) + $P_B \le 1.5\ S_m$ d) P_m(or P_L) + P_B + $Q \le 3.0\ S_m$ 2) Upset a) $P_m < S_m$, b) P_L S 1.5 S_m(SIC) c) P_m(or P_L) + P_B + $P_B \le 1.5\ S_m$ d) P_m(or P_L) + P_B + $Q \le 3.0\ S_m$ 3) Faulted i) $P_m \le 1.2\ S_m$ or S_y whichever is larger, AND P_m(or P_L) + $P_B \le 1.5\ (1.2)\ S_m$ or $1.5\ S_y$ whichever is larger ii) Table 5.2-15 p. 3.7-30, p. 5.2-46, T 5.2-15	ANSI B31.7-1969 ASME BPVC Sec. III p. 3.1-101 p. 3.7-49

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
NOT AVAILABLE	Analysis and testing p. 3.10-1	NOT AVAILABLE	IEEE Standard 344-1971 p. 3.10-1

SEISMIC REVIEW TABLE

Docket Number
50-269, 270, 287

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Oconee Nuclear Station Unit Nos. 1,2,3 Reactor type: PWR Containment type: 6 buttresses with shallow dome (pre-stressed concrete) NSSS Manufacturer: Babcock & Wilcox Architect Engineer: Utility & Bechtel Unit #1: 11-67/2-73 Unit #2: 11-67/10-73 Unit #3: 11-67/7-74	0.05 for rock foundation Sec. 2.6 p. 2-9	0.03	VI	0.10 for rock foundation. 0.15 for overburden foundation. Sec. 2.6 p. 2-9	0.07	Time history record of the N-S, May 1940 El Centro Earthquake was used (vertical and N-S horizontal components) at 0.01 sec intervals for the first 30 sec of duration. Sec. 1C.3.4.2.1 p. 1C-4d	3 components: Each horizontal combined with the vertical simultaneously. Sec. 5A.2.2 p. 5A-3	Absolute sum p.5-19	R-S smooth curve with max. acceleration of .15g @ 2% damping. Housner. Plate II-4 App. 2B	Time-history method. Sec. 1C.3.4.2.1 p. 1C-4d Sec. 1C.3.4.2.2(b) p. 1C-4e-4f

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION: TYPE	BEARING INFORMATION: THICKNESS	BEARING INFORMATION: V_s PROFILE	GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
Reinforced concrete foundation slab. Depth = 8½ feet thick. Founded on bedrock.	Banded biotite hornblande geniss and granite geniss. The surface has weathered unevenly and the residual soils grade down irregularly.	Sound Rock is found at depths of 5-40 feet	Not available in FSAR	Not available in FSAR.	"Design of Keowee and Jocassee Dam" Refer to PSAR p. 2.4.3 and Question 8.6-PSAR Supp. 1, Question 12.1-PSAR Supp. 4, Question 12.2-PSAR Supp. 4, Item 11-PSAR Supp. 5, Item 1-PSAR Supp. 6.	Stick model with soil springs.	Not available	in FSAR.	2% OBE 5% SSE
Sec. 5.1.2.1 p. 5-2			Refer to Sec. 2.5 and Sec. 2.6 p. 2-8 in PSAR	Refer to PSAR 2.4.4 Sec. 2.4.5 p. 2-8	Sec. 2.4.4 p. 2-8	Sec. 5.1.3.2 p. 5-18	Refer to Sec. 2.5, p. 2-8, in PSAR	and Sec. 2.6 2-9	p. 5-12 Fig. 5-10

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="4">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING
OBE/SSE</th><th rowspan="2">(% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td>Welded carbon and stainless steel assemblies:

Steel framed structures:

Reinforced concrete equipment supports:

Reinforced concrete frames and buildings:

Prestressed concrete structures
(i) under design earthquake forces
(ii) under maximum hypothetical earthquake

Sec. 5A.2.2
p. 5A-3</td><td>1.0

2.0

2.0

5.0

2.0

5.0</td><td>$\underline{Y}=1/\phi(1.0D+1.0P+1.0T+E')$
$\underline{Y}=1/\phi(1.05D+1.25P+1.0T+1.25E$ or $W)$
$\underline{Y}=1/\phi(1.05D+1.5P+1.0T)$
$\underline{Y}=1/\phi(1.0D+1.0W_t+1.0P_i)$ for tornado forces
$\underline{Y}$=required yield strength of structure
D=dead loads
P=design accident pressure
T=thermal load
E=seismic load based on design earthquake
E'=seismic load based on maximum hypothetical earthquake
W=wind load
P_i=stress due to differential pressure
ϕ=capacity reduction factor

For further details refer to
Sec. 5A.2.2, p. 5A-2</td><td>ACI 318-63

ACI 301

ASME, PVBC, Sec. III, VIII, IX

Sec. 5.1.2.1
p. 5-4</td></tr>
</table>

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="5">MECHANICAL & PIPING</th></tr>
<tr><th rowspan="2">DAMPING
OBE/SSE</th><th rowspan="2">(% critical damping)</th><th rowspan="2">METHOD
OF
QUALIFICATION</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td>Vital piping:

Sec. 5A.2.2
p. 5A-3</td><td>0.5</td><td>Analytical

Sec. 1C.3.4.1
p. 1C-4ai</td><td>(A) piping:
I. Design loads + design earthquake loads
$P_m \le 1.0S_m$
$P_L+P_b \le 1.5S_m$
II. Design loads + maximum hypothetical earthquake loads
$P_m \le 1.2S_m$
$P_L+P_b \le 1.2(1.5S_m)$
III. Design loads + pipe rupture loads
$P_m \le 1.2S_m$
$P_L+P_b \le 1.2(1.5S_m)$
IV. Design loads + maximum hypothetical earthquake loads + pipe rupture loads
$P_m \le 2/3S_u$
$P_L+P_b \le 2/3S_u$

P_L=Primary local membrane stress intensity
P_b=Primary bending stress intensity
P_m=Primary general membrane stress intensity
S_m=Allowable membrane stress intensity
S_u=Ultimate stress

p. 4-4</td><td><u>For piping:</u>
Nuclear power piping code USAS B31.7, Sec. 1C.3, p. 1C-3

<u>Mechanical components:</u>
-ASME, Sec. III for nuclear vessels.
-S_m values Table N-421 of ASME code.

Sec. 4.1.2.5.1
Sec. 4.1.2.5.2
p. 4-3</td></tr>
</table>

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and tests. Table 8.8 p. 8-36	Not available.	No detailed information available. Refer to Table 8.8 for some seismic considerations. p. 8-36

SEISMIC REVIEW TABLE *

Docket Number
50-219

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Oyster Creek Nuclear Power Station Unit 1 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Burns & Roe, Inc. 12-64/4-69	0.11 Sec. V.3 p. V-3-1	0.073 Sec. V.3 p. V-3-5	VII	0.22 Sec. V.3 p. V-3-5	0.147 Sec. V.3 p. V-3 -5	Not used	2 components: Horizontal and vertical added directly and linearly for: Reactor building. Control room/turbine building, rad. waste building. Horizontal only for intake structure. Sect. 3.6 Amend. 11 Q IV-3-1	SRSS Amend.11 Quest. IV-2-1	Housner spectra used for analysis of reactor building, ventilation stack, control room, rad-waste bldg. Equivalent static method for intake structure, suction header, spent fuel pool Question IV. 2 Amend. 11, Sec. V-3-1.2, FDSAR, Sec. 3.5.1	No floor response spectra: Seismic Design curves for FWCI piping and equipment. p. 5-11, 12 Amend. 38

*Information from BNL Docket search and SEPB Report "Seismic Review of Oyster Creek Nuclear Power Plant for SEP", Phase I Report.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Mat foundation Grade: + 23 ft MSL Foundation: -11 ft MSL	Fine to medium texture sand of med. density	17 ft.	Not available	Wells are 60 to 70 ft. or more in depth.	Not available	Rocking mode analyzed separately in seismic analysis of reactor and control room/ turbine building. Using a torsional spring to represent the foundation flexibility.	Not available	Not available	Not available
	alternating layers of clay, silt and fine sand	17 ft							
	dense sand, med. to coarse texture	65 ft.							
	layers of clay, silt and fine sand	8 ft.							
	dense fine to coarse sand	below							
Sec. II.5.2	Sec. II.5.2	Sec. II.5.2		Sec. II.4, p. II-4-1					

SEISMIC REVIEW TABLE

STRUCTURES				
DAMPING OBE/SSE (% critical damping)		DESIGN CRITERIA		
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES	
			Reinforcing Steel Max. Tension	Concrete Max. Allowable Compression
Reinforced concrete structures (reactor building)	10.0	Reactor building., Control Room., Battery Room., Intake Structure.* 1. DL + LL + OL + E (0.11g) 2. DL + LL + OL + W 3. DL + LL + OL + E´(0.22g)	1. 0.5 F_y 2. 0.667 F_y 3. 0.90 F_y	0.45 f'_c 0.60 f'_c 0.90 f'_c
steel frame structures	2.0	Reactor Concrete Pedestel** 1. DL + equipment + jet load + temperature + OBE 2. DL + equipment + jet load + temperature + SSE	1. 0.25 F_y 2. 0.25 F_y	0.133 f'_c (bending) 0.267 f'_c (bending)
welded assemblies	1.0	Drywell Concrete Shield*** 1. DL + LL + over pressure + max. temp. + OBE 2. DL + LL + over pressure + max. temp. + SSE 3. DL + LL + max. temp. + OBE + jet force	1. 0.50 F_y 2. 0.50 F_y 3. 0.667 F_y	0.45 f'_c 0.45 f'_c 0.60 f'_c
bolted and riveted assemblies	2.0			
reinforced concrete stack	5.0			
Sec. V.3, p. Table V-3-1				

*Table V-3-3, Table 1-A-4, Amend. 22
**Table 1-A-2, Amend. 22
***Table 1-A-1, Amend. 22

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
		LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Bolted and riveted assemblies 2.0 2. Welded assemblies 1.0 3. Vital piping 0.5 Table V-3-1 Sec. V.3 p. V-3-2	Not available	Class I piping* Thermal MOL + SL MOL + 2(SL) MOL = Max. operating loads SL = Seismic loads due to OBE $S_A = f(1.25\ S_c + 0.25\ S_H)$ f = stress range reduction factor S_c, S_H = a-lowable stress, ASA B31.1	Allowable stress S_A S_H Safe shutdown can be achieved	See load combinations and Supplement 6, Amend. 68, Appendix 6.
		Reactor vessel supports* Seismic - Seismic + jet - 2(seismic) -	Normal AISC allowables 150% of normal AISC allowables 150% of normal AISC allowables	
		Primary containment ** DL + operating + LOCA + E DL + operating + LOCA + E′ * Ques. IV. 1, Amend. 11 ** Table V-3-2, Sec. 3.8.1	ASME Sec. VIII Code case 1272N-5	

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE (% Critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Quoted from answer to Question IV.1, Amend 11 "The control room panels and auxiliary racks are usually shipped assembled and therefore these units must be designed for normal shipping shock which is in the order of several g's acceleration. Certain components are removed and padded to reduce vibration effect and excessive acceleration. In all cases, however, the design analysis is made of the panels and instruments. All relays in safety circuits are energized; and since they are capable of closing against 1.0g, they can certainly maintain contact during an acceleration of 0.22g." Question IV.1, Amend. 11	

SEISMIC REVIEW TABLE*

Docket Number
50-255

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Palisades Nuclear Generating Plant Unit 1 Reactor type: PWR Containment type: 6 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Combustion Engineering Architect Engineer: Bechtel	0.10	0.067	VII	0.20	0.13	Containment design Housner spectra. For floor response spectra generation and for equipment and piping the 1952 TAFT earthquake was used, whose R-S envelops the Housner spectra.	Maximum horizontal component with vertical component simultaneously.	SRSS - response spectra method for structural modes and piping.	Housner design spectrum	Not clear - it appears that TAFT 1952 earthquake was used to generate floor response spectra. Then from lumped-mass model, the accelerations at each floor level were obtained and the TAFT response spectra were scaled to those valves. Static method used for piping with frequency > 20 Hz. For vertical R-S, 2/3 of horizontal ground spectrum Ref. 3.Q.5.8 and Q.5.6.
3-67/3-71	p. 2-16	Sec. A.2 p. A-7		p. 2-16	Sec. A.2 p. A-7		Sec. A.2 p. A-7		Question 5.13 p. 5.13-1	

*Information obtained from BNL Docket search and SEPB Report, "Seismic Review of Palisades NPP Unit No. 1".

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete slab 8 1/2 to 13 ft. thick Sec. 5.1.2 p. 5-2	Loose dune sand overlies about 30 ft. of well-compacted, gray silty sand. Below this is about 90 ft. of compact till. Bedrock, Mississippian Coldwater Shale, is reached at a depth of about 150 ft. below	lake level. It is composed of blue, gray or red shale.**	5400 fps for lake deposits 6700 fps for glacial till 10,000 fps for bedrock	10 ft. from ground surface Sec. 2.4.1, p. 2-14, p. 5.10-21	Not available	Containment: Lumped mass, spring model. Determine horizontal spring constant and 2 vertical springs which provide rotational restraint. "Building FNDT. interaction effects". 10-66, ASCE Engr. Mech.	Not available	Not available	Not available

Sec. 2.3.1
p. 2-10 to p. 2-11

** Type and thickness of bearing information are presented together.

SEISMIC REVIEW TABLE

STRUCTURES

DAMPING OBE/SSE (% critical damping)		DESIGN CRITERIA: LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Welded steel framed structures 2. Bolted steel framed structures 3. Reinforced concrete: structures on soil including structural damping 4. Prestressed concrete: containment structure on soil including structural damping	2.0/2.0 2.0/2.0 5.0/7.5 4.0/7.5	*Final design (SSE) for Class I structures except the containment shell* 1. $Y = 1/\phi\,(1.25D + 1.0R + 1.25E)$ 2. $Y = 1/\phi\,(1.25D + 1.25H + 1.25E)$ 3. $Y = 1/\phi\,(1.25D + 1.25H + 1.25E)$ (0.9 D is used where dead load subtracts from critical stress in the above two equations) 4. $Y = 1/\phi\,(1.0D + 1.0\,R + 1.0E')$ 5. $Y = 1/\phi\,(1.0D + 1.0H + 1.0E')$	ACI 318-63 Code Utlimate strength design Sec. A.2, p. A-3, Appendix A
		Final design (SSE) of the containment structure $(.7 < \phi < .9)$ a) $Y = 1/\phi\,(1.05D + 1.5P + 1.0\,T_A + 1.0F)$ b) $Y = 1/\phi\,(1.05D + 1.25P + 1.0T_A + 1.25H + 1.25E + 1.0F)$ c) $Y = 1/\phi\,(1.05D + 1.25H + 1.0R + 1.0F + 1.25E + 1.0T_o)$ d) $Y = 1/\phi\,(1.05D + 1.0F + 1.25H + 1.25W + 1.0\,T_o)$ e) $Y = 1/\phi\,(1.0D + 1.0P + 1.0T_A + 1.0H + 1.0E' + 1.0T_o)$ f) $Y = 1/\phi\,(1.0D + 1.0H + 1.0R + 1.0E' + 1.0F + 1.0\,T_o)$	Sec. B.1.6 p. B-5, Appendix B
Sec. A.2, p. A-8, Appendix A		Y = Required yield strength of the structures D = Dead load of structure and equipment + any other permanent loads contributing stress, such as soil or hydrostatic loads R = Force or pressure on structure due to rupture of any one pipe H = Force on structure due to thermal expansion of pipes under operating conditions. E = *Design seismic load for Class I structures* E′ = Maximum seismic load for Class I structures W = Wind load for Class I structures, tornado load for containment ϕ = Capacity reduction factor (Defined in B.1.7) P = Design accident pressure loads F = Effective prestress loads T_o = Thermal loads due to temperature gradient through wall during operating conditions T_A = Thermal loads due to temperature gradient through the wall and expansion	*Containment Working Stress:* a. $D + L + F + T_o$ b. $D + L + F + T_A + E$ (or W) c. $P' = 1.15P$ FSAR App. B.1

SEISMIC REVIEW TABLE

MECHANICAL & PIPING

DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1) Welded steel plate assemblies	1.0/1.0	Analytical method	<u>Critical reactor vessel internal structural</u> 1. Design loading + design earthquake forces $P_m \le S_m$, $P_B + P_L \le 1.5 S_m$	P_L, P_m, S_m, S_y are defined in the ASME Boiler and Pressure Vessel Codes, Section III, Article 4.
2) Concrete equipment supports on another structures	2.0/2.0	DC control centers 250V-test	2. Normal operating loadings + hypothetical earthquake forces $P_m \le S_D$ $P_B \le 1.5 [1-(\frac{P_m}{S_D})^2] S_D$	ASA B31.1 "USA Standard Code for pressure piping power piping." Piping: FSAR App. A Q.5.12, Q.5.7
3) Steel piping	0.5/0.5		3. Normal operating loadings + hypothetical earthquake forces + pipe rupture loadings $P_m \le S_L$ $P_B \le 1.5 [1-(\frac{P_m}{S_L})^2] S_L$	
			S_u = Minimum tensil strength of material at temperature $S_L = S_y + (1/3)\ (S_u - S_y)$ S_D = Design stress = 1.2 S_m	Sec. 3.2 p. 3.6
Sec. A.2 Appendix A		Question 5.8 p. 5.8-3	<u>Class 1 systems and equipment design (including piping)</u> 1. MOL + PTT + SL — 1. Applicable code allowable stress 2. MOL + MTT + SL — 2. Minimum yield stress at temperature 3. MOL + MTT + 2SL — 3. Minimum yield stress at temperature	may be exceed but limited to no more than + 10%

MOL = Maximum normal operating load including design pressure, design temperature + piping and support reactions

PTT = Normal planned thermal transients associated with expected plant normal operating transients such as start-up, shutdown and load swings

MTT = Maximum thermal transients in the systems functioning during plant energency conditions such as full power reactor trip turbine generator trip, loss of auxiliary power and the DBA

SL = Design seismic load resulting from a seismic ground surface acceleration of 0.1g

2SL = Hypothetical seismic load resulting from a seismic ground surface acceleration of 0.2g

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-277, 278

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Peach Bottom Atomic Power Station, Unit 2 and 3 Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Bechtel Unit 2: 1-68/8-73 Unit 3: 1-68/7-74	0.05 p.C.2-1	0.033 p.C.2-2	VII Sec. 2.5. 3.1.1, p.2.5-12	0.12 p.C.2-2	0.08 p.C.2-2	Synthetic time-history.	2 components H+V simultaneous p. C.4-1 Sec. C.2.2 Sec. C.3.3	Absolute sum (Response spectrum analysis) Sec. C.3.3 p. C.3-3	Housner OBE: Fig. C.3.1 SSE: Fig. C.3.2 Max. acceleration = 0.15g @ 2% damping p. C.3-2,	Time-history method using an earthquake time-history whose raw spectrum response curve is greater than or equal to the site design response spectrum curve. Sec. C.3.3 p. C.3-3

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Class I structures: Spread or mat foundation on fresh rock Peters Creek Schist. Depth: Not available Auxiliary building: Steel H bearing pile foundation.	Residual soils. Weathered Peters Creek Schist. Fresh Peters Creek Schist.	0 to 40 ft. below surface. 25 to 65 ft. in thickness. 15 to 80 ft. below surface.	Not available	Varies from 12 to 15 ft. near and upstream. Reaches 100 ft. one mile downstream.	Site is 9 miles above Conowingo Dam; 6 miles below Holtwood Dam	Fig. C.3.3 indicates fixed base stick model.	Not available	Not available	Not available
p. 2.7-3, p. 2.7.4	p. 2.5-14	p.2.5-14		p. 2.5-10	p. 2.5-10	p. C.3.3			

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td>Reinforced concrete strutures 2.0/5.0
Steel framed structures 2.0/5.0
Weld steel assemblies 1.0/2.0
Bolted and riveted assemblies 2.0/5.0

p. C.2-2</td><td>1. D + E
2. D + E´
3. D + W
4. D + W´
5. D + E + T
6. D + E´ + T
7. D + F
where D = Dead load; W = Wind load; W´ = Tornado load; E = OBE; E´ = DBE; T = Thermal; F = Flood

p. C.2-6
p. C.2-7
For further reference, refer Appendix C</td><td>AISC for structural steel
ACI 318-63 for reinforced concrete
<u>Maximum allowable stresses</u>
Steel - .9 yield strength
Concrete - .85 compressive strength
Reinforcement- .9 yield strength

See Codes on p. C.2-8.

p. C.2-6</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Welded steel assemblies 1.0/2.0 Bolted and riveted assemblies 2.0/5.0 Seismic Class I Piping System 0.5/0.5 p. C.2-2	Analysis and tests p. C.5-1	Normal and upset: 1. D. W. + pressure 2. D. W. + pressure + OBE 3. D. W. + pressure + thermal 4. D. W. + pressure + OBE + thermal Emergency: 1. D. W. + DBE Faulted: 1. D. W. + DBE + Jet reaction forces For further details refer to TAble C.5.6, Table C.5.7	Reactor Vessel ASME BPVC III Piping USAS B 31.1.0 Table C.5.6 Table C.5.7

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Test and empirical experience. p. C.5-1	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-293

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA: OBE HOR. g	EARTHQUAKE DATA: OBE VERT. g	EARTHQUAKE DATA: SSE INTENSITY MM	EARTHQUAKE DATA: SSE HOR. g	EARTHQUAKE DATA: SSE VERT. g	EARTHQUAKE DATA: EARTHQUAKE TIME HISTORY	METHOD OF COMBINATION: NO. OF EARTH. COMP. USED AND ITS COMB.	METHOD OF COMBINATION: MODAL COMB.	DESIGN SPECTRA: TYPE OF GROUND DESIGN SPECTRA	DESIGN SPECTRA: METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
Pilgrim Nuclear Power Station Unit No. 1 Reactor type: BWR Containment type: Reinforced Concrete NSSS Manufacturer: General Electric Architect Engineer: Bechtel	0.08	0.053	VII	0.15	0.10	Taft Earthquake, July 21, 1952 normalized to 0.08g and 0.15g ground accelerate was used for computer analysis and results compared against those from smothed response spectra method.	Horizontal component and vertical component were acting simultaneously.	For piping system: SRSS For struct: not available.	Housner	Time-history method using Taft record. Then each curve was compared to the ground response spectrum and corrected to fall below the ground spectrum curve.
8-68/6-72	Sec. 2.5.3.2 p. 2.5-6	App. C, Sec. C.2.2 p. C.0-1	Sec. 2.5.3.2 p. 2.5-6	Sec. 2.5.3.2 p. 2.5-6	App. C, Sec. C.2.2 p. C.0-1	Sec. 12.2.3.5.2 p. 12.2-5	Comment 12.2.4 p. 2-26	App. C, Sec. C.3.3 p. C.0-7	Fig. 2.5-5 Fig. 2.5-6	Sec. 12.2.3.5.2, p. 12.2-6 Comment: 12.2.2 p. 2-22

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Heavily rein- forced concrete mat 8 ft. depth Sec. 12.2.2.1, p. 12.2-2	Layers of glacial and recent deposits. Upper layer sandy silts (about 20 ft.) lower layer (glacial zone) poorly graded to well graded sands with varying a- mount of gravel. Boulders are scattered thru-out the soils. Bedrock is Dedham Granodiorite Group with a depth of six feet. Sec. 2.5.2.4.2 and Sec. 2.5.2.4.3 p. 2.5-4		Not available.	Ground water table generally follows the site topo- graphy. i.e., moderately steep ground water gra- dients are present with flow toward Cape Cod Bay. Water level is a- bout 2 1/2 to 5 ft. from surface (gathered from boring logs). Sec. 2.4.1.3.2, p. 2.4-1	Not avail- able.	Stick model with soil springs. Sec. 12.2.3.5.2 p. 12.2-5	Not available.	Not available.	Not avail- able.

SEISMIC REVIEW TABLE

STRUCTURES		DESIGN CRITERIA	
	DAMPING OBE/SSE (% critical damping)	**LOAD COMBINATION**	**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Reinforced concrete building	5.0/7.5	1. Dead load + OBE.	1. Stresses according AISC. and ACI Codes.
Internal concrete structures and equipment supports	2.0/3.0	2. Dead load + wind loading.	2. Maximum allowable stress increased 1/3 above normal code-allowable stress.
Steel frame structures	2.0/5.0		
Bolted steel assemblies	2.0/5.0	3. Dead load + jet forces and pressure and temperature transient with rupture of single pipe + OBE.	3. Normal code-allowable stress.
Welded assemblies	1.0/2.0	4. Dead load + R + SSE	4. Steel - 15% of AISC Code allowable stress concrete -0.75 f'_c where "working stress design" method is used. Reinforcement = 0.9 f_y when "ultimate strength design" method is used. Load factor of 9.0 is used with appropiate reduction factor as in ACI-318-63.
		R= Jet forces and pressure and temperature transient with rupture of single pipe.	
Table 12.2.3, p. 12.2-6		Details: See C.2.3, App. C, p. C.0-2	Details: See C.2.3, App. C, p. C.0-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Class I Piping System 0.5/1.0	Both analytical and empirical (testing).	Load combinations are presented as tables. Per ASME Code. Drywell membrane stresses: D + R + E — stress intensities are defined per code paragraph N-413 and their limits as per code N-413. D + R + flood	ASME BPVC Section III
Table 12.2.3, p. 12.2-6 Table 12.2.3-2	App. C, C.3.1, p. C.0-5	Table C-9 Table C-20	Sec. C.3.4, App. C, p. C.0-7

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number
50-266, 301

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY mm	HOR. g	VERT. g					
Point Beach Nuclear Plant Unit No. 1 & 2 Reactor type: PWR Containment type: 6 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Bechtel	0.06	0.04	NOT AVAILABLE	0.18	0.08	NOT AVAILABLE	Horizontal & Vertical Components Combined Simultaneously	SRSS	Housner Spectra	Olympia, Washington N80E on April 13, 1949 Earthquake normalized to .06g was used for this analysis.
Unit 1: 7-67/10-70 Unit 2: 7-68/11-71	Sec. 5.1 p. 5.1-41			Sec. 5.1 p. 5.1-41			Append. A p. A-3	Sec. 5.1.2.4 p. 5.1-52	Q. 5.2 p. 5.2-2 Fig. A-1 & A-2	Append. A. p. A-18

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
For containment building: Reinforced concrete foundation slab which is supported by steel H-piles Depth is not available Sec. 1.2 p. 1.2-2 Sec. 2.11.4 p. 2.11-3	Overburden soils: silty clay, silty sand, sand, gravel, cobbles and boulders Bedrock: Niagara dolomite the bedrock as a whole consists of dolomites limestones and sandstones. Sec. 2.9.3 p. 2.9-2	70 ft. to 100 ft. NOT AVAILABLE	NOT AVAILABLE	"The potable water for use at the Point Beach Plant is drawn from a 257 ft. deep well." Sec. 2.6 p. 2.6-10	NOT AVAILABLE	Structure: Stick Model Soil: Cantilever Beam assumption indicates fixed base modelling Q.5.15 p.Q5.15-6	NOT AVAILABLE	OBE/SSE: 5.0/5.0 % of damping factors. Append. A p. A-5	NOT AVAILABLE

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING
OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td>Welded Steel Framed Structures 2.0/2.0
Bolted Steel Framed Structures 2.5/5.0
Reinforced Concrete Structures on Soils 5.0/7.5
Prestressed Concrete Containment Structures on Piles 2.0/5.0</td><td>For Containment Structures:
a) $Y = 1/\phi\ (1.05D + 1.5p + 1.0TA + 1.0F)$
b) $Y = 1/\phi\ (1.05D + 1.25p + 1.0TA + 1.25H = 1.25E + 1.0F)$
c) $Y = 1/\phi\ (1.05D + 1.25H + 1.0R + 1.0F + 1.25E + 1.0To)$
d) $Y = 1/\phi\ (1.05D + 1.0F + 1.25H + 1.0W + 1.0To)$
e) $Y = 1/\phi\ (1.0D + 1.0p + 1.0TA + 1.0H + 1.0E' + 1.0F)$
f) $Y = 1/\phi\ (1.0D + 1.0H + 1.0R + 1.0E' + 1.0F + 1.0To)$
Note: 0.95D is used instead of 1.05D where dead load subtracts critical stress.</td><td>For Concrete Structures of the Reactor Containment:
ACI-318-63.
For further details refer to Sec. 5.1 p. 5.1-8</td></tr>
<tr><td>Append. A
p. A-5
Table A.1-1</td><td>Sec. 5.1
p. 5.1-26</td><td>Sec. 5.1
p. 5.1-2</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE (% critical damping)		**METHOD OF QUALIFICATION**	**DESIGN CRITERIA**			
			LOAD COMBINATION			**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
				Pressure Vessel	Piping	
Interior Concrete Equip. Supports	2.0/2.0	Analytical & Testing	Normal Conditions	(a) $P_m \le S_m$		For pressure piping: ASME BPVC, USAS B31.3
Vital Piping Systems	0.5/0.5			(b) $P_m(\text{or } P_L)+P_B \le 1.5S_m$		
Welded Steel Plate Assemblies	1.0/2.0			(c) $P_m(\text{or } P_L)+P_B+Q \le 3.0S_m$	$P \le S$	For reactor vessel: ASME Sec. III, Class A
			Upset Conditions (Normal + OBE)	(a) $P_m \le S_m$		
				(b) $P_m(\text{or } P_L)+P_B \le 1.5S_m$		
				(c) $P_m(\text{or } P_L)+P_B+Q \le 3.0S_m$	$P \le 1.2S$	
			Emergency Conditions	(a) $P_m \le 1.2S_m$ or $P_m \le S_y$ whichever is larger		
				(b) $P_m(\text{or } P_L)+P_B \le 1.5\ (1.2S_m)$ or $P_m(\text{or } P_L)+P_B \le 1.5\ (S_y)$ whichever is larger	$P \le 1.2S$	
			Faulted Conditions (Normal + DBE, Normal + DBA, Normal + DBE + DBA')	Design Limit Curves of WCAP-5890, Rev. 1 as Modified by Note 1 of This Appendix	Same as Pressure Vessel	
Append. A p. A-5 Table A.1-1		Append. A p. A-3 & Vol. 2 Sec. 5.1 p. 5.1-41	P_m = Primary general membrane stress intensity P_L = Primary local membrane stress intensity P_B = Primary bending stress intensity Q = Secondary stress intensity			Append. A p. A-3 Sec. 4 Table 4.1-9

S_m = Stress intensity from ASME BPVC III Code

S_y = Minimum specified material yield

DBA' = Steady-state Portion of Design Basis Acciden

Append. A, Table A.3-1

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
NOT AVAILABLE	Testing as per WCAP 7397-1. Q.5.2 p. 5.2-2	NOT AVAILABLE	NOT AVAILABLE

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Plant grade is at Elv. 693.5 1. Mat foundation at Elv. 674 App. A-1, p. 5.12 Because of problem with liquefaction of soils above Elv. 645 due to ground acceleration, the soil above Elv. 645 is densified to a minimum relative density of 85%. For further details refer to App. Al, Sec. 5	Overburden materials are permeable sandy Alluvial soils from glacial outwash and recent river deposits. The bedrock is sandstone of Franconia formation of < 180 feet in thickness.	Structures are founded on densified sandy Alluvial soils of 158 to 185 feet. Sec. 2.9.4 p. 2.9-4&2.9-5	Elv. 0-20 loose sand 470 ft/sec Elv. 20-50 med dense 2150 ft/sec Elv. 50-180 very dense 2860 ft/sec Elv. 180-4100 Sandstone 5020 ft/sec App. A Sec. 4 Plate 4.1	Ground water table is 5 ft to 20 ft of the ground surface of the site and slope southwest from the Missippi River toward Vermillion River. Sec. 2.7.1 p. 2.7-1	Lock and dam number 2 is 17 miles upstream of plant site. It is 3250 ft long dike, 2 single-lift locks with chambers 110'x600' and 110'x 500'; and a spillway section of 20-30 ft. Amend. 22 Sec. 2.7.3 p. 2.7-8C	Stick model with soil springs. App. B Sec. B6.3 p. 3.6-6	Not available.	5% of critical damping. Amend. 12 App. B Table B.6-5	Not available.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="5">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING OBE/SSE</th><th rowspan="2">(% critical damping)</th><th colspan="3">DESIGN CRITERIA</th></tr>
<tr><th colspan="2">LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr><td></td><td></td><td>L. C.</td><td>Class 1</td><td>R/C: Steel</td></tr>
<tr><td>Reactor building containment vessel:</td><td>1.0/1.0</td><td>Normal operating</td><td>D + L + (W or S)</td><td>ACI 318-63 / AISC</td></tr>
<tr><td>Reactor building shield structure:</td><td>2.0/2.0</td><td>OBE</td><td>D + L + DBA + greater of the OBE + (W or S)</td><td>" / "</td></tr>
<tr><td>Reactor building internal concrete construction:</td><td>5.0/5.0</td><td>DBE</td><td>D + L + S + DBA + DBE</td><td>1 1/2 times ACI 318-63
1 1/2 AISC</td></tr>
<tr><td>Steel framed structures:</td><td>2.0/2.0</td><td>Tornado</td><td>D + L + tornado + tornado missiles</td><td>$f_c = 0.85 f_c'$ $f_s = 0.9 F_y$
$f_s = 0.9 F_y$</td></tr>
<tr><td>Reinforced concrete construction:</td><td>2.0/2.0</td><td></td><td></td><td></td></tr>
<tr><td></td><td></td><td>Other</td><td>Jet forces, rupture loads, flood whereever applicable</td><td></td></tr>
<tr><td colspan="2">Amend. 12 (11-15-71)
App. B
Table B.6-5</td><td colspan="2">For details refer to
App. B, Sec. B.6.1, p. B.6-1 and
Table B.6-1.</td><td>App. B
Sec. B.3
p. B.3-1</td></tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE	(% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION			DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
				Vessel	Piping	
Piping systems:	0.5/0.5	Analytical and testing.	1. Normal condition (D.L. thermal and pressure)	(a) $P_m \le S_m$ (b) P_m (or P_L)+$P_B \le 1.5S_m$ (c) P_m (or P_L)+P_B+Q $\le 3.0S_m$	$P \le S$	ASME, BPVC, Section III ANSI B31.1, 1967 (App. B., Table B.7-3) p. 5.2-11 App. B Table B.7-3
Mechanical equipment:	2.0/2.0		2. Upset condition (normal and OBE)	(a) $P_m \le S_m$ (b) P_m (or P_L)+$P_B \le 1.5S_m$ (c) P_m (or P_L)+P_B+Q $\le 3.0S_m$	$P \le 1.2S$	LOAD COMBINATION(cont.)
			3. Emergency condition	(a) $P_m \le 1.2S_m$ or S_y whichever is larger (b) P_m (or P_L)+$P_B \le 1.5(1.2S_m)$ or $1.5S_y$ whichever is larger	$P \le 1.5(1.2S)$	P_m=Primary general membrane stress intensity P_L=Primary local membrane stress intensity P_B=Primary bending stress intensity Q =Secondary stress intensity
			4. Faulted condition (Normal+DBE+pipe rupture)	(a) $P_m \le 1.5S_m$ or $1.2S_y$ whichever is larger (b) P_m (or P_L)+$P_B \le 2.25S_m$ or $1.875S_y$ whichever is larger	$P \le S_y$ or 1.8S	S_m=Allowable stress intensity value from ASME, BPVC S_y=Maximum specified material yield strength Amend. 24 (10-6-72) Table 5.2-1
Amend. 12 (11-15-71) App. B Table B.6-5		App. B Sec. B.7(i) p. B.7-9 p. B.7-14	S_y=Minimum specified yield strength (ASME, BPVC Code, Sec. III) Amend. 11, App. B., Table B.7-2 and Table B.7-3			P =Stress S =Allowable stress from ANSI B31.1 code for power piping 1967

App. B, Table B.7-3.
p. 5.2-11

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-254, 265

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Quad - Cities Station Unit 1 and 2 Reactor type: BWR Containment type: Mark I(Steel) NSSS Manufacturer: General Electric Architect Engineer: Sargent & Lundy, Engineers	0.12	0.08	VII	0.24	0.16	South-East component of San Francisco Golden Gate 1952 earthquake normalized to a maximum ground acceleration.	Horizontal and vertical components combined simultaneously.	SRSS	Ground response spectra for the Golden Gate Park earthquake as well as the Housner spectra.	Normalized Golden Gate 1952 earthquake was used for the Time History Method.
Unit 1: 2-67/9-71 Unit 2: 2-67/3-72	Sec. 2.6 p. 2.6-1	Sec. 12.1.1.3 p. 12.1-6	Sec. 12.1.1.3 p. 2.6-1	Sec. 2.6 p. 2.6-1	Sec. 12.1.1.3 p.12.1-6	Append. C p. E-1	Sec. 12.1.2 p.12.1-9	Sec. 12.1.2 p.12.1-9	Amend. 13, Sec. 12, p. 12.1-1, Fig. 12.1-1	Sec. 12, Amend.13 p. 12.3-8

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reactor building: Reinforced concrete foundation. 297 ft.-0 by 150 ft. -0	Zone 1: (above El 530) contains both good & poor zones much of bad rock has been excavated	0 to 20 ft.	Turbine Room No. 1. Middle Grout Zone 8,000 to 9,000 fps. above Upper Soft Zone 5,500 to 7,500 fps. Upper Soft Zone 3,900 to 5,100 fps. Good Rock Zone 8,000 fps. Lower Soft Zone 4,700 to 6,200 fps. below Deep Soft Zone 6,000 fps.	Not available.	This site is about midway between Lock and Dam No. 14 and 13 on Mississippi River.	Structure: Stick Model Soil: Fig. 12.1.6 shows fixed base assumption.	300,000 psi to 1,500,000 psi	Not available.	Not available.
	Zone 2: (El 530-500) primarily good rock	30 ft.							
	Zone 3: (El 500-475) both good & poor zones.	25 ft.							
	Zone 4: (Below El 475) primarily good rock	50 ft.							
Sec. 12.1.2.1 p. 12.1-7	Amend. 15 p. 13 Table 4		Amend. 15, p.6		Sec. 2.4, p. 2.4-1	Sec. 12.1.2, p. 12.1-8 and p. 12.1-9	Amend. 15, 1 of 2		

SEISMIC REVIEW TABLE

STRUCTURES		
DAMPING OBE/SSE (% critical damping)	DESIGN CRITERIA	
	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reinforced concrete structure 5.0* Steel frame structure 2.0 Welded assemblies 1.0 Bolted and riveted assemblies 2.0 *(For both O.B.E and D.B.E.) Sec. 12.1.1.3, Table 12.1.1, p. 12.1-6 and p. 12.2-4	Primary containment (including penetrations) = a) D + P + H + T + E b) D + P + H + T + E c) D + P + H + T + E´ Class I structure = D + R + E D + R + E´ D + L D = Dead load; L = Wind live load P = Pressure due to loss-of-coolant accident R = Jet force or pressure on structure due to rupture of any one pipe H = Force on structure due to thermal expansion of pipes under operating conditions T = Thermal loads on containment, reactor vessel, and internals due to loss-of-coolant accident. E = Design earthquake load, ground horizontal g = 0.12, vertical g = 0.68 E´ = Maximum earthquake load, ground horizontal g = 0.24, vertical g = 0.16 Amend. Sec. 12, p. 12.1-3 ∿ p. 12.1-6	AISC - For structure steel ACI - 318 - 63 Amend. 13, Sec. 12, p. 12.13-1

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
		DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital Piping Systems 0.5 (For both O.B.E. and D.B.E. except for the standby gas treatment system, where 1% of critical damping was used).	Analytical	Reactor primary vessel supports = a) D + H + E b) D + H + R + E c) D + H + E′ Reactor primary vessel internals = a) D + E b) D + E′ c) P + D + T Other major Class I equipment = a) D + T + M + E b) D + T + M + E′ For designations refer to previous page.	For reactor pressure vessel: ASME Boil and Pressure Code, Sec. III, 1963 and Summer 1964, Append. A. Class I piping: USAS B31.1
Sec. 12.1.1.3, Table 12.1.1 p. 12.1-6	Amend. Sec. 12 p. 12.2-14	Amend. 13, Sec. 12, p. 12.3-10	Append. C p. ii, Amend. Sec. 12, p. 12.1-4

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number
50-312

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Rancho Seco Nuclear Generating Station Unit No. 1 Reactor type: PWR Containment type: 3 buttresses with shallow dome (pre-stressed concrete) NSSS Manufacturer: Babcock and Wilcox Architect Engineer: Bechtel	0.13	0.09	VI	0.25	0.17	1952 Taft Earthquake	Three earthquake components: two horizontal and one vertical. Results for each horizontal earthquake were added separately on absolute basis to those from vertical earthquake; yielding two distinct seismic loading cases.	SRSS both for structures and piping.	Accelerogram of Taft Earthquake 1952. The response spectra are broadened in their range of peak responses.	Time-history acceleration method
10-68/8-74	p. 5.1-2	p. 5.1-2		p. 5.1-2	p. 5.1-2	Appendix 5B p. 5B-4	Question AEC 5.51 p. 5A-51	Question AEC 5-51 p. 5A-51	Appendix 5B p. 5B-4, Figs. SK6292-S-59 and SK6292-S-62	Appendix B p. 5B-4

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION: TYPE	BEARING INFORMATION: THICKNESS	BEARING INFORMATION: V_s PROFILE	GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
Circular reinforced concrete mat 8 ft thickness Foundation is found about 35 ft below grade surface. Sec. 5.2.1 p. 5.2-1 Appendix 2E p. 2E-1	The granite & metamorphic basement is overlain in the site by 1500 to 2000 ft tertiary or older sediments. The surface unit is pliocene laguna formation of firm siltstone, sand, gravel Appendix 2C, p 2C-7, Table 2C-1.2 Sec. 2.5	The surface unit of pliocene laguna formation is about 126 ft.	Not available.	150 ft below original ground surface. p. 2.4-1	1. Data on reservoirs and lakes within 50-mile radius are given in Table 2.4-1. 2. Plot of on-site dam, Question AEC No. 2.14. Appendix 2A p. 2A-132	Stick model with soil springs. Sec. 5.2.1.3.6 p. 5.2-18	Not available	10% for design basis earthquake. Appendix 5B p. 5B-6	Not available

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Stress level:* 1. a) Welded structural steel, reinforced or prestressed concrete, no cracking, no joint slip. — 0.5/1.0 2. a) Welded structural steel, reinforced and prestressed concrete (only slight cracking). — 2.0 b) Reinforced concrete with considerable cracking. — 3.0/5.0 c) Bolted and/or riveted steel. — 5.0/7.0 3. a) Welded structural steel, prestressed concrete (without complete loss in prestress). — 5.0 b) Prestressed concrete with no prestress left. — 7.0 c) Reinforced concrete. — 7.0/10.0 d) Bolted and/or riveted steel. — 10.0/15.0 4. Rocking of entire structure — 5.0/9.0 Translation of entire structure — 30 (OBE, SSE) *NOTE. Stress level 1 = low, well below proportional limit. Stress below 1/4 yield point. Stress level 2 = Working stress Stress level 3 = At or just below yield point Stress level 4 = Varies Appendix 5B p. 5B-7	A) $\phi C = (1\pm0.05)\ D + 1.5\ P + 1.0\ T_A + 1.0\ F$ B) $\phi C = (1\pm0.05)\ D + 1.25\ P + 1.0\ T_A + 1.25\ H + 1.25\ E + 1.0\ F$ C) $\phi C = (1\pm0.05)\ D + 1.25\ H + 1.0\ R + 1.0\ F + 1.25\ E + 1.0\ T_c$ D) $\phi C = 1.0\ D + 1.0\ P + 1.0\ T_A + 1.0\ E' + 1.0\ F + 1.0\ H$ E) $\phi C = 1.0\ D + 1.0\ H + 1.0\ R + 1.0\ E' + 1.0\ F + 1.0\ T_o$ ϕ = Capacity reduction factor. D = Dead loads of structures and equipment plus any other permanent loading contribution stress, such as hydrostatic or soil. P = Design accident pressure load. F = Effective prestress loads. R = Force or pressure on structure due to rupture of any one pipe. H = Force on structure due to thermal expansion or contraction of pipes due to design conditions. T_o = Thermal loads due to the temperature gradient during operating conditions. T_A = Thermal loads due to the temperature gradient. E = OBE C = Required capacity to resist factored loads. E' = DBE p. 5.1-6 and p. 5.2-7	1. ACI-318-63 Ultimate strength method Question AEC 5.23, p. 5A-25 2. AISC (Sixth Edition) Sec. 5.1.3, p. 5.1-4 NOTE: 1. Normal working stress. Design methods are used for design load case. 2. Factored load case--to check the capacity to withstand accident conditions. Sec. 5.1.4, p. 5.1-4a For details see: Sec. 5.2.1.3 p. 5.2-11

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital piping systems or equipment. Low, well below proportional limit, stress below 1/4 yield point. 0.5 Working stress, no more than point. 0.5/1.0 At or just below point. 0.5/2.0 p. 5B-7	Dynamic analysis Testing Question AEC 5.49 p. 5A-49	I. Design loads + OBE loads: $P_m \leq 1.0\ S_m$; $P_L+P_B \leq 1.5\ S_m$ II. Design loads + DBE loads: $P_m \leq 1.2\ S_m$; $P_L+P_B \leq 1.2\ (1.5\ S_m)$ III. Design loads plus pipe rupture load: $S_m \leq 2/3\ S_U$; $P_L+P_B \leq 2/3\ S_U$ IV. Design loads + DBE + pipe + rupture loads: $P_m \leq 2/3\ S_U$; $P_L+P_B \leq 2/3\ S_U$ P_L = Primary local membrane stress intensity. P_m = Primary general membrane stress intensity. P_B = Primary bending stress intensity. S_m = Allowable membrane stress intensity. S_U = Ultimate stress for unirradiated material at operating temperature. p. 4.1-4	Nuclear vessels: ASME BPVC 1967, Section III Piping: USAS I, B31.7 p. 4.1-5

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Test data/or calculations of equipment to with-stand OBE and DBE are provided by vendors. Question AEC 5.67 p. 5A-62	Not available.	Not available.

SEISMIC REVIEW TABLE*

Docket Number
50-244

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Robert Emmett Ginna Nuclear Power Plant, Unit No. 1 Reactor type: PWR Containment type: cylindrical without buttresses (prestressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Gilbert 4-66/9-69	0.08 Sec. 5.1.2.4 p. 5.1.2-15	0.08 Sec. 5.1.2.4 p. 5.1.2-15	V Sec. 2.9 p. 2.9-1	0.20 Sec. 5.1.2.4 p. 5.1.2-15	0.20 Sec. 5.1.2.4 p. 5.1.2-15	None used	Two comp., larger horizontal plus vertical, combined via "direct addition" vertical component is assumed unamplified due to high axial stiffness of the containment.	None (Containment analyzed as single degree of freedom).	Housner	Equivalent static approach based on Housner ground spectra. Multimode response spectrum analysis used to check containment vessel and RHRS pipeline from RCS loop to containment.

*Information was obtained from BNL Docket Search and SEPB Report "Seismic Review of Ginna Nuclear Power Station Unit No. 1 for SEP, Phase 1 Report".

SEISMIC REVIEW TABLE*

Docket Number
50-244

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Robert Emmett Ginna Nuclear Power Plant, Unit No. 1 Reactor type: PWR Containment type: cylindrical without buttresses (prestressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Gilbert 4-66/9-69	0.08 Sec. 5.1.2.4 p. 5.1.2-15	0.08 Sec. 5.1.2.4 p. 5.1.2-15	V Sec. 2.9 p. 2.9-1	0.20 Sec. 5.1.2.4 p. 5.1.2-15	0.20 Sec. 5.1.2.4 p. 5.1.2-15	None used	Two comp., larger horizontal plus vertical, combined via "direct addition" vertical component is assumed unamplified due to high axial stiffness of the containment.	None (Containment analyzed as single degree of freedom).	Housner	Equivalent static approach based on Housner ground spectra. Multimode response spectrum analysis used to check containment vessel and RHRS pipeline from RCS loop to containment.

*Information was obtained from BNL Docket Search and SEPB Report "Seismic Review of Ginna Nuclear Power Station Unit No. 1 for SEP, Phase 1 Report".

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="4">STRUCTURES</th></tr>
<tr><th colspan="2" rowspan="2">DAMPING
OBE/SSE (% critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</th></tr>
<tr>
<td>1. Containment structure (prestressed cylindrical wall)

2. Concrete support structure for reactor vessel and steam generator

3. Steel assemblies
a) Bolted or riveted
b) Welded

4. Other concrete above ground

Table 5.1.2-1</td>
<td>2.0

2.0

2.5
1.0

5.0</td>
<td>Containment Structure Loading Combinations:
Normal- 12 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_S + 2.0\ E$

Test- 4 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_W + 1.15\ IP$

Accident Pressure-
Cond. "d" - 12 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_W + 1.0\ IP + 1.0\ AT_{60} + 0.8\ E$ (a=0.1g)

Cond. "a" - 4 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_W + 1.5\ IP + 1.0\ AT_{90}$

Cond. "b" -8 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_W + 1.25\ IP + 1.0\ AT_{90} + E$

Cond. "c" - 8 load combinations, example
$1.0\ DL + 1.17\ VP + 1.0\ OT_S + 1.0\ IP + 1.0\ AT_{60} + 2.0\ E$

DL = Dead load
VP = Vertical prestress
$OT_{W,S}$ = Operating temp. winter, summer
App. 5D, Table 5.1.2-41 FSAR
IP = Internal pressure (p=60 psig)
AT_{60} = Accident pressure + temperature (p = 60 psig, T = 286°F)
E = Design earthquake (a=0.1g)</td>
<td>ACI-318
AISC - 63
State of New York Building Construction Code, 1961 (Class III structures)

5.1.2.3
FSAR 5.1.2.4, 7.2</td>
</tr>
</table>

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Vital Piping System	0.5	Analytical and Testing	Loading Combination / Vessels and Reactor Internals / Piping 1. Normal + OBE: $P_m \leq S_m$; $P_L + P_B \leq 1.5\ S_m$ / $P_m \leq 1.2\ S$; $P_L + P_B \leq 1.2\ S$ 2. Normal + SSE: $P_m \leq 1.2\ S_m$; $P_L + P_B \leq 1.2(1.5\ S_m)$ / $P_m \leq 1.2\ S$; $P_L + P_B \leq 1.2(1.5\ S)$ 3. Normal + Pipe rupture loads: $P_m \leq 1.2\ S_m$; $P_L + P_B \leq 1.2\ (1.5\ S_m)$ / $P_m \leq 1.2\ S$; $P_L + P_B \leq 1.2(1.5\ S)$ P_m = Primary general membrane stress; or stress intensity P_L = Primary local membrane stress; or stress intensity P_B = Primary bending stress; or stress intensity S_m = Stress intensity value from ASME B and PV Code Sec. III S = Allowable stress from USAS B31.1 Code for pressure piping	ASME BPVC Sec. III, USAS B31.1 Supports -- Working stress -- within yield after load redistribution -- within yield after load redistribution Fuel Pool Racks: Reg. guides 1.13, 26, 28, 38, 60, 61 ANSI N 18.2 - 1973 ANSI N 45.2.2 - 1972 ANSI N 45.2.13 - 1974 Structural Welding Code AWS Spec. D1.1 Rev. 2-74 ASME BPV Code, Sec. III, Sec. VIII, and IX, 1974 AISC - 1974 FSAR 9.5, App. 14A
Table 5.1.2-1		Amend. 2, Question 5	FSAR 5.1.2, FSAR App. 4-A, Table 1	

Equipment: FSAR Table 3.2.3-2 through 3.2.3-7

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Testing Amend. 2 Question 5	Class I instrumentation: Control Room: Racks have been assembled and the mounting and wiring of all components has been designed such that the functions of the circuits or equipment will perform in accordance with prescribed limits when subjected to seismic accelerations of 0.21g in the horizontal and vertical direction simultaneously. Control room, containment, and auxiliary bldg: Mounting and wiring of all components has been done such that simultaneous accelerations of 0.52g in the horizontal and vertical planes will not dislodge, cause relative movement or result in any loss or change of function of circuits or equipment. Section 5.1.2.4, 7.2	Not available

SEISMIC REVIEW TABLE

Docket Number
50-272, 311

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Salem Nuclear Generating Station Units 1 and 2 New Jersey Reactor type: PWR Containment type: Atmospheric (reinforced concrete) NSSS Manufacturer: Westinghouse Architect Engineer: United Engineers and Constructors	0.10	0.067	VII	0.20	.133	El Centro (N-S) May 18, 1940 normalized to 0.10g to 0.20g for OBE and DBE respectively was used for containment structure analysis by step by step integration method.	The vertical component was considered to be acting simultaneously with the horizontal motion.	1.Response spectra analysis: Sq root of sum of squares but if < 3 modes→ absolute sum of maximum values. 2.Time history analysis (finite element method): summing of significant modes.	1. For freq > 0.33 cps: Aug spectra developed by Housner. 2. For freq < 0.33 cps: Utilized data suggested by Newmark. Fig. IIC-3a Fig. IIC-3b	Time history method.
Unit #1: 12-66/8-71 Unit #2: 10-67/8-71	Sec. 2.9 p. 2.9-1	Sec. 2.9 p. 2.9-1	Sec. 2.9 p. 2.9-1	Sec. 2.9 p. 2.9-1	Sec. 2.9 p. 2.9-1	Sec. 5.2.4.2 p. 5.2-17	Sec. 5.2.4.2 p. 5.2-17	App. C Sec. C.3.3 p. C.3-2	App. B p. IIC-10	App. C Sec. C.3.3 p. C.3-2

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Circular concrete mat Depth 16 ft Sec. 5.6.2 See Table 5.6-1	1800 feet of sediments. Upper 35 feet includes hydraulic fill and Quaternary alluvium of clay silt and some sand and gravel. Vincentown formation is encountered at about 70 feet. App. B p. IIC-9	Foundations are established directly in Paleocene silty sands of Vincentown formation or upon compacted fill extended to Vincentown. Depth of Vincentown is 90 feet. App. B p. IIC-9	3500 ft./sec App. B. p. IIC-9	Water level is about 20' from surface, but ground water movement thru acquifiers is quite low due to low permeability. The direction is going into Delaware River. App. B p. IIB-14 Table IIB-2	Not available.	Two methods were used: 1. Lumped mass model analysis using aug resp. spectra 2. Finite element modal analysis, for structure and soil. The most conservative results are used. Sec. 5.2.4.2 p. 5.2-17	Not available.	2%—OBE 5%--DBE Sec. 5.2.4.2 p. 5.2-17	Not available.

See Plate IIC-1, App. B

SEISMIC REVIEW TABLE

STRUCTURES			
		DESIGN CRITERIA	
DAMPING OBE/SSE	(% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Concrete structures: Structural steel: Bolted or riveted Welded App. C Sec. C.3.2 p. C.3-1	2.0/5.0 2.5 1.0	1. Operating + DBA + OBE $C=1.0D\pm0.05D+1.25P+1.0(T'+TL')+1.25E+1.0B$ 2. Operating + DBA + DBE $C=1.0D\pm0.05D+1.0P+1.0(T''+TL'')+1.0E'+1.0B$ 3. Operating + DBE $C=1.0D\pm0.05D+1.0T'''+1.0E'+1.0B$ C = Required load capacity of section D = Dead load P = Accident pressure load T' = Load due to maximum temperature gradient based upon temperature associated with 1.25 times accident pressure T'' = Load due to maximum temperature gradient based upon temperature associated with accident pressure T''' = Load due to operating temperature gradient thru the steel liner, concrete shell and mat E = Load from OBE E' = Load from DBE TL' = Load exerted by liner based upon temperature with 1.25 times accident pressure TL'' = Load exerted by liner based upon temperature associated with accident pressure. Sec. 5.2.3 p. 5.2-7 to 5.2-8	ACI 318-63 AISC Manual, 6th edition Note: (a) For normal operating + OBE" "Working Stress Design" ACI 318-63 and the allowable stresses are 1/3 above the normal applicable code working stresses. (b) For normal load + DBE: "Ultimate Strength Design" ACI 318-63 Sec. 5.6.3 p. 5.6-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
		DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital piping system: 0.5 App. C Sec. C.3.2 p. C.3-1	Not available.	Vessel / Piping 1. Normal condition: Vessel: (a) $P_m \leq S_m$ (b) P_m or $(P_L)+P_b \leq 1.5S_m$ (c) $P_m(P_L)+P_b+Q \leq 3.0S_m$ Piping: (a) $P_m \leq S$ (b) P_m or $(P_L)+P_b \leq S$ 2. Upset condition: Vessel: (a) $P_m \leq S_m$ (b) $P_m(P_L)+P_b \leq 1.5S_m$ (c) $P_m(P_L)+P_b+Q \leq 3.0S_m$ Piping: (a) $P_m \leq 1.2S$ (b) P_m or $(P_L)+P_b \leq 1.5(1.2S)$ 3. Emergency condition: Vessel: (a) $P_m \leq 1.2S_m$ or S_y whichever is larger (b) $P_m(P_L)+P_b \leq 1.5(1.2S_m)$ or $1.5S_y$ whichever is larger Piping: (a) $P_m \leq 1.2S$ (b) P_m or $(P_L)+P_b \leq 1.5(1.2S)$ 4. Faulted condition: Vessel: Design limit curves* Piping: Design limit curves* NOTE: P_m = primary general membrane stress, P_L = primary local membrane stress, P_b = primary bending stress, S_m = stress value for ASME, BPVC code, Section III, nuclear vessels, S_y = minimum specified material yield, S = allowable stress from USASI, B31.1 code for press piping. App. C, Table C.4-2	ASME Nuclear Vessel Code Section III ANSI B31.1 for piping Sec. 5.2.8.3 p. 5.2-53

*Design limit curves developed using 50% of ultimate strain as maximum allowable membrane strain.

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE*

Docket Number
50-206

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
San Onofre Nuclear Generating Station Unit 1 Reactor type: PWR Containment type: Dry containment-spherical (steel) NSSS Manufacturer: Westinghouse Architect Engineer: Bechtel 3-64/3-67	0.25 0.25g for Cat. A, 0.20g for Cat. B, UBC for Cat. C.	0.167	Not available	0.50 0.50g for initial design, 0.67g for reevaluation of reactor bldg, NSSS, containment sphere, concrete sphere enclosure, foundation and cradle and diesel gen. bldg.	0.33 (0.44g for re-evaluation)	A synthetic time history was generated so that it's response spectra envelop the Housner spectra at 2% damping. 3.7.1-1	(see below)	(see below)	Housner spectra used in original design and 1972-75 re-evaluation except that a site specific spectra was used for the concrete sphere enclosure and the deisel generator bldg. 3.7.1.1	Floor response spectra by time history method for re-evaluation of RCL piping, equipment, and NSSS supports. All other Category "A" piping and equipment (ECCS, SIS, feedwater lines, CVC) - 1.0g and 0.67g for horizontal and vertical, 0.5g, Housner spectra for equipment.

System	Analysis	E.Q.Comp.&Comb.	~~Modal~~ Comb.
Reactor bldg.	Res. Spec.	3comp. R.G. 1.92	R.G. 1.92
D.G. Bldg.	Res. Spec.	3comp. R.G. 1.92	R.G. 1.92
Steel Containment sphere	Res. Spec.	3comp. SRSS	SRSS
RCL piping and equipment	Time history	3comp. algebraic	Direct integration
RCL supports	Time history	3comp. algebraic	Direct integration
Concrete sphere enclosure	Res. Spec	3comp. SRSS	SRSS
Diesel Gen. Bldg.	Res. Spec.	3comp. R.G. 1.92	R.G.1.92
Main bldg.	Res. Spec.	3comp. Absolute	Absolute
Intake struct., Aux. bldg., battery rm.	Res. Spec.	2comp. Absolute	N/A

*Information from BNL Docket search and SEPB Report No. EDAC-175-166.01, August '79, "Seismic Design Bases and Criteria for San Onofre Nuclear Generating Station, Unit 1".

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
The foundation of the containment-reactor building is in the shape of a spherical segment extending from the ground surface to a depth of 40 ft. The supporting medium is San Mateo sand. Bedrock is at a depth of 1000 ft. p. 3.7.1-6	1. Quarternary terrance deposits: Consist of clayey or silty, fine to coarse sand with some cobbles. Average thickness is 40 ft. 2. San Mateo formation: Massive, well-graded, fine	to coarse sand with gravel and occasional lenses of thin beded gray shale or siltstone. Approx. 1000 ft. thick.	Surface terrace deposit 400 and 1250 fps San Mateo sand 765 fps Capistrano siltstones 2000 fps Monterey shale 2160 fps San Onofre Breccia 5000 fps undifferentiated sediments 3,900 fps	Average level of ground water is 15 ft. below original grade (El + 5ft. MLLW Datum), and the gradient is 17 ft. per mile toward the ocean. Sec. 1.1.4 p. 1-10	Not available	Soil-structure interaction is represented by a set of six frequency-independent interaction springs attached to the reactor building structure at the center of gravity of the base mat. "SHAKE" program used.	Not available	Soil horizontal translation - 12% Soil vertical transalation - 18% Soil rocking - 10% These values include radiation and material damping.	Not available
	Sec. 1.3.2 p. 1-56					p. 3.7.2-4		Table 3.7.1-3	

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Reactor vessel internals (stainless steel core support structure) (a) welded assemblies (b) bolted assemblies 2. Reinforced concrete reactor support 3. Steel containment vessel and foundation 4. Framed steel structures 5. Concrete structures above ground (a) shear wall type (b) rigid frame type Sec. 9.2.2 Table 9.1 p. 9-10	 1.0 2.0 4.0 4.0 2.5 7.0 5.0	<u>Concrete structures</u> (concrete sphere enclosure) $u = 1.4\ D + 1.7\ L$ $u = 1.4\ D + 1.7\ L + 1.9\ E$ $u = D + L + T_o + R_o + E'$ $u = D + L + T_A + R_A + 1.0\ P_A + (Y_R + Y_j + Y_M) + E'$ (9 additional L.C.) <u>Steel structures</u> $S = D + L$ $1.6S = D + L + T_o + R_o + E'$ $1.6S = D + L + T_A^o + R_A^o + P_A + 1.0\ (Y_R + Y_j + Y_M) + E$ (8 additional L.C.) <u>Reactor building and foundation and cradle support</u> $u = 1.0\ D + 1.0\ L + 1.0$ (DBE) $0.9Y = 1.0\ D + 1.0\ L + 1.0$ (DBE) Y = ultimate strength of section	Concrete sphere enclosure, Reactor bldg. (concrete internals), foundation and cradle, diesel generation bldg. - ACI 318 - 71, AISC 1971 Main building, intake structures, auxiliary bldg., battery rm., turbine pedestal - ACI 318 - 63 - AISC 1963 - UBC 1964 Refueling Water Stg. Tank-API publication for storage tank.

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1. Vital Piping Systems 0.5 Sec. 9.2.2 Table 9.1 p. 9-10	Not Available	Containment sphere Primary membrane and bending stresses are evaluated at: A). Basic shell thickness under combined dead weight, design pressure and seismic loads B). Shell to base mat juncture under combined deadweight, design pressure and seismic loads. C). Shell in vicinity of equipment hatch and personnel lock D). Main feedwater penetration under combined dead weight, internal pressure, seismic, and piping.	Containment sphere: ASME Sect. III, 1971 and 1972 Summer Addenda Allowable Stress / Ref. Primary Membrane: S_M / NE-3221.1 Primary Membrane plus primary bending: $1.5S_M$ / NE-3221.3 Primary plus secondary: $3.0S_M$ / NE-3131.0, NB-3222.2 Equipment&Piping,RCL&NSSS Supports Category A, ASME, Section III, 1971, NB-3600;All other Category A piping and equipment (feedwater, CVC, ECCS, ACS): ASME Section III - 1962, USAS B 31.1 (1964) DIESEL Gen-IEEE-STD-344

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Tested or evaluated to determine that the instruments would withstand 1.0g without misoperation. Amend. 10, Suppl. 1, Quest. 14	Not available	Not available

SEISMIC REVIEW TABLE

Docket Number

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Shippingport Project 129 Reactor type: PWR Containment type: Dry containment-spherical (steel) NSSS Manufacturer: Westinghouse Architect Engineer: Burns and Roe, Inc. also Stone and Webster Engineering Corp.	← Not available →									

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
← Not Available →									

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	ASME Code Sec. VIII 1952 Ed. P.A. Regulations for pressure vessels 1954 ed.

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not Available			

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not Available			

SEISMIC REVIEW TABLE

Docket Number
50-335

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
St. Lucie Plant, Unit No. 1. Reactor type: PWR Containment type: Dry containment-cylindrical (steel) NSSS Manufacturer: Combustion Engineering Architect Engineer: Ebasco 7-70/3-76	0.05 Sec. 2.5 p. 2.5-25a	0.033 for shield building Sec. 3.8. 2.2, p. 3.8-67, Amend. 32	VI Sec. 2.5 p. 2.5-27	0.10	0.067 for shield building Sec. 3.8. 2.2, p. 3.8-67, Amend. 32	Synthetic time-history Sec. 2.5.3 p. 2.5-28	Each horizontal combined with the vertical on an absolute sum basis. Resulting two load cases. Sec. 3.7. 3.2.4, p. 3.7-43a	SRSS Sec. 3.7. 2, p. 3.7-19	Housner spectra Fig. 2.5-23 and 24 Fig. 3.7-1 and 2	Time-history method using synthetic time history Sec. 3.7 p. 3.7-36 Sec. 3.7 p. 3.7-3 Rev. 16

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
For reactor building: Rigid foundation mat.	Loose sand with small amounts of silt and clay, containing isolated pockets of shell fragments and limestone nodules.	50' to 60'	Not available	Shallow non-artesian aquifer extends to a depth of about 150 ft. below land surface	"No dams are located within the hydrological influence of Hutchinson Island."	Stick model with soil springs.	Generally utilize shear shear moduli ranging from ranging from 16,700 psi to 14,000 psi.	Not available	Not available
p. 2.5-1	More dense contains a greater percentage of fines (material finer than the no. 200 sieve) and has very few pockets of limestone nodules and *	60' to 150'	Sec. 2.5-36 p. 2.5-38	Vol. 1, Sec. 2.4 p. 2.4-20	Sec. 2.4, p. 2.4-7a	Sec. 3.7.2.1.1 p. 3.7-6	Sec. 2.5, p. 2.5-38	p. 2.5-19	

(Note: Due to space the columns for Type and Depth had to be continued here....)

* shell fragments
more clayey than the material above, does not contain pockets of shells and limestone, and is dense in consistency. Vol. 1, Sec. 2.5, p. 2.5-8 — 150' to at least 400'

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE (% critical damping)		DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
		Shield Building	
Welded steel framed structure	2.0/2.0	(1.0 ± 0.05)(D + T) + 1.25 LOCA + 1.25 OBE (1.0 ± 0.05)(D + T) + 1.25 OBE (1.0 ± 0.05)(D + T) + 1.0 LOCA + 1.0 DBE (1.0 ± 0.05)(D + T) + 1.0 DBE	AIC 318-63 Φ Yield capacity reduction factors are used. Sec. 3.8.2.2.8 p. 3.8-71
Bolted or riveted steel framed structure	2.5/2.5		
Reinforced concrete frames and buildings	2.0/5.0		
Steel containment vessel	2.0/2.0	For further details refer to Sec. 3.8.2.2 p. 3.8-68 of Amend. 32-9/6/74.	AISC -1969 Sec. 3.8 p. 3.8-2,3
Sec. 3.7 p. 3.7-3a			

SEISMIC REVIEW TABLE

MECHANICAL & PIPING				
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA		
		LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Welded steel plate assemblies 1.0/1.0 Reinforced concrete equipment supports 2.0/5.0 Steel piping 0.5/0.5 Sec. 3.7 p. 3.7-3a	Analytical and Testing Sec. 3.7, p. 3.7-36,43a Sec. 3.9 p. 3.9-1	<u>Containment Vessel</u> LOCA + OBE: $P_M \leq 1.0\ S_m$ $P_L + P_B \leq 1.5\ S_m$ $P_L + P_B + Q \leq 3.0\ S_m$ LOCA + DBE: $P_M \leq 0.9\ S_y$ $P_L + P_B \leq 0.9\ S_u$ OBE + Pipe rupture: $P_M \leq 1.0\ S_m$ $P_L + P_B \leq 1.5\ S_m$ $P_L + P_B + Q \leq 3.0\ S_m$ DBE + Pipe rupture: $P_M \leq 0.9\ S_y$ $P_L + P_B \leq 0.9\ S_u$ OBE + Thermal + Seismic loads on piping: $P_M \leq 1.0\ S_m$ $P_L + P_B \leq 1.5\ S_m$ $P_L + P_B + Q \leq 3.0\ S_m$ LOCA + DBE with pressure & thermal + seismic loads on piping: $P_M \leq 0.9\ S_y$ $P_L + P_B \leq 0.9\ S_u$ Table 3.8-7 p. 3.8-30 Amend. 32	<u>Piping</u> Design: $P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$ $P_L + P_b + P_e + Q \leq 3.0\ S_m$ Normal: $P_L + P_b + P_e + Q + F = S_p$ (use fatigue curve) Upset: (Press + Wt. + OBE + VT) $P_L + P_b + P_e + Q \leq 3.0 S_m$ $P_L + P_b + P_e + Q + F = S_p$ (use fatigue curve) Emergency: (Press + Wt. + DBE) Max press $\leq$ 1.5 design press $P_L + P_b \leq 2.25\ S_m$ Faulted: (Press + Wt. + DBE + rupture) Max press $\leq$ 2.0 design press. $P_L + P_b \leq 3.0\ S_m$ Table 3.9-3 p. 3.9-18 Amend. 38	ASME BPVC Sec. III Sec. 3.8 p. 3.8-14 Rev. 13, 7-15-73 ANSIB31.7 Sec. 3.9 Table 3.9-3 p. 3.9-18

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Testing and Inspection Sec. 8.3 p. 8.3-23	Type II - 600 v penetration assembly. A steel plate barrier has been erected inside the containment in the electrical system penetrations: $D + P_R \leq$ 90 percent of material yield strength $D + OBE \leq$ normal AISC working stress $D + DBE \leq$ 90 percent of material yield strength Vol. 2 Sec. 3.8 p. 3.8-33, Rev. 15 (10-11-73)	IEEE - 317, April 1971 Standard for electrical assemblies in containment structure for nuclear fueled power generating stations. Sec. 3.8, p. 3.8-33, Rev. 15, 10-11-73. IEEE -279 (Aug. 1968) IEEE -308 (Nov. 1970) Sec. 8.1, p. 8.1-2

SEISMIC REVIEW TABLE

Docket Number

50-280, 281

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY mm	HOR. g	VERT. g					
Surry Power Station Unit 1 & 2 Reactor type: PWR Containment type: sub-atmospheric (reinforced concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Stone and Webster	0.07	0.046	VII	0.15	0.10	Synthetic time-history	For Class 1 Structures Hor. & Vert. Combined simultaneously	SRSS	1) For frequencies higher than 2 cycles/sec. Housner Spectra 2) Frequency range between 0.3 cycles/sec. Housner Average Spectra have been normalized to a max. ground velocity of about 4"/sec for O.B.E. and 9"/sec for P.B.E. 3) For frequencies lower than about 0.3 cycles/sec. using data suggested by Dr. Newmark & Hall.	The floor response spectra are encompassed by the umbrella spectrum used in the dynamic analyses of Westinghouse supplied equipment. RCL analysis done with floor response spectra
Unit 1: 6-68/5-72 Unit 2: 6-68/1-73	Sec. 2.5.4 p. 2.5.4-1 2-13-70		Sec. 2.5 p. 2.5.5-5 2-13-70	Sec. 2.5.5 p. 2.5.5-1 p. 2.5.5-7 12-1-69		Q4.23, Supp. 1	p. 15.2-16 B.1-1	Sup. Vol. 1 2.4.10 p.S4.10-2 10-15-70 & Sup. Vol.1 Q.4.12 p. S4.12 p. S4.12-2 10-15-70	Sec. 2.5.5 p. 2.5.5-9 Fig. 2.5-4, 2.5-5	App. B, p. B.3-I Supp. Vol. 1 Q.4.10, Q 5.10, 4.12

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
For Major Class I Struct. (except for the fuel building and main steam valve enclosure struct.): MAT Foundation. 10 ft. thick Pile foundation: turbine foundation, spent fuel pit, mainsteam shielding, RWST	Surface Deposits -Consist of sand, silty sand thin layers of iron oxide-cemented sands and clays of Norfolk Estuarine Formation	50' 80'	Not available Sec. 2.5 p. 2.5.5-2 TYPE - THICKNESS (cont.) * Below this lie formations of — Thickness Eocene 45' Paleocene 55' Cretaceous 800'	18 wells within a five mile radius of the site. Depth from 280'∿799' 4 operating water wells on the site obtain water from the Eocene sediments at depth about 400'	NOT AVAILABLE	STICK MODEL with soil springs	NOT AVAILABLE	O.B.E/S.S.E. 0.05/0.10 This is an overall value which includes the damping in both the reinforced concrete structure and the damping.	O.B.E/S.S.E 0.02/0.05
	Below this lies clays, compact sand and silt members, and shell fragments of the Chesapeake Formation	240' thickness varying from -16 msl to -47 msl *	Crystalline Bedrock. — Estimated at a depth of about 1300'						
Sec. 15.4 p. 15.4-8 Sec. 15.5 p. 15.5.1-1 p. 2.4.6-1			Sec. 2.4 p. 2.4.2-2	Part B Vol. 1 Sec. 2.3		p. 15.5.1.4-2 Append. B Sec. B.2 p. B.3-1		Sec. 15.5 p. 15.5.1.4-2 & p. 15.5.1.4-3	Supp. Vol.1 Q. 5.22 p. S5.22-1

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE (% of Crit. Damping)		DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
1) Containment Struct. & Foundation	5.0/10.0	1. Operating + DBA = (1.0$\pm$0.05) D+1.5P+1.0 (T+TL) 2. Operating + DBA + OBE = (1.0$\pm$0.05) D+1.0P+1.0($\underline{T}$+$\underline{TL}$) +1.5E 3. Operating + DBA + DBE = (1.0$\pm$0.05) D+1.0P+1.0($\underline{T}$+$\underline{TL}$) +1.0HE 4. Operating + 1.25DBA + 1.25OBE = (1.0$\pm$0.05) D + (1.25p)+(T'+TL') +1.25E 5. Operating + Tornado Loading = (1.0$\pm$0.05) D + 1.0$\underline{T}$' + 1.0C	For Containment Struct. ACI 318-63 Part IV-B
2) Steel Framed Struct. Including Supporting Struct. and Foundations			
a) Bolted	2.5		
b) Welded	1.0		
3) Concrete Struct. Aboveground			
a) Shear-wall type	5.0		
b) Rigid-frame type	5.0		
Sec. 15.3 & p. 15.5.1.4-3 Sec. 15.2 Table 15.2.4-1 p. 15.2-19	Supp. Vol. 1 Q. 5.12 P. S5.12-1	Sec. 15.5 Table 15.5.1.2-1 p. 15.5.1.2-4 4-15-70	Sec. 15.5 p. 15.5.1.2-2

SEISMIC REVIEW TABLE

MECHANICAL & PIPING

DAMPING OBE/SSE (% Critical Damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA: LOAD COMBINATION	PRESSURE VESSELS	PIPINGS	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reactor Vessel Internals of Control Rod Assembly Drives: a) Welded assemblies b) Bolted assemblies	 1.0 2.0	Analytical & Testing	Normal Conditions:	$P_m \le Sm$ $P_m(\text{or } P_L)+P_B \le 1.5S_m$ $P_m(\text{or } P_L)+PB+Q \le 3.0S_m$	$P_m \le S$	ASME BPVC SEC. III USAS B31.1
Vital Piping Systems: a) Carbon steel b) Stainless steel	 0.5/1.0 0.5/1.0		Upset Conditions:	$P_m \le S_m$ $P_m(\text{or } P_L)+P_B \le 1.5S_m$ $P_m(\text{or } P_L)+P_B+Q \le 3.0S_m$	$P_m \le 1.2S$	
Reinforced concrete reactor support structure including the reactor vessel	5.0					
Mechanical equipment, including pumps, fans, and similar items	2.0		Emergency Conditions:	$P_m \le 1.2S_m$ or $P_m \le S_y$ whichever is larger $P_m(\text{or } P_L)+P_B \le 1.5(1.2S_m)$ or $P_m(\text{or } P_L)+P_B \le 1.5(S_y)$ whichever is larger	$P_m \le 1.2S$	
			Faulted Conditions:	Design Limit Curves of WCAP-5890	Design Limit Curves of WCAP-5890	
Sec. 15.2 Table 15.2.4-1 p. 15.2-19 Supp. Vol. 1 Q5.12 p. S5.12-1		Sec. B.5 p. b.5-1 Table B.5-1 Supp. Vol. 1 Q 4.10 p. S4.10-1	P_m = Primary general membrane stress intensity P_L = Primary local membrane stress intensity P_B = Primary bending stress intensity Q = Secondary stress intensity			App. B p. B.2-8 p. B.2-10 p. B.2-13

S_m = Stress intensity value from ASME BPVC III
S_y = Minimum specified material yield
For further details refer to App. B, Talbe B.2-1, p. B.2-6

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
NOT AVAILABLE	Tests Method. (This tests data is contained in WCAP-7397-L Seismic Testing of Electrical and Control Systems Equipment) Supp. Vol. 1 Q.4.11 p. S4.11-1 3-15-71	NOT AVAILABLE	NOT AVAILABLE

SEISMIC REVIEW TABLE

Docket Number
50-289

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Three Mile Island Unit 1 Reactor type: PWR Containment type: 6 buttresses with shallow dome (prestressed concrete) NSSS Manufacturer: Babcock and Wilcox Architect Engineer: Gilbert	0.06	0.04	VI	0.12	0.08	1957 Golden Gate Park - Average smooth revised with 1940 El Centro - normalized to ground acceleration of 0.06g. Synthetic time-history for floor response spectra	Horizontal and vertical combined by absolute sum.	Piping: SRSS and modes 10% within each other are added absolutely.	Actual spectra envelops Golden Gate and El Centro earthquake time histories.	Time-history method. Gilbert Topical Report # 1729 "Dynamic Analysis of Vital Piping Systems Subjected to Seismic Motion."
5-68/4-74	Sec. 5.1.2.1.1 p. 5-10	Sec. 5.1.2.1.1 p. 5-10	Sec. 2.8.1 p. 2-41	Sec. 5.1.2.1.1 p. 5-10	Sec. 5.1.2.1.1 p. 5-10	Sec. 2.7.1, p. 2-31 Sec. 2.8.2, p. 2-42	Sec. 5.2.4.1.2 p. 5-52	Sec. 5.4.5.1 p. 5-76a p. 5-52	Sec. 2-7, p. 2-31 Fig. 2-24 Fig 5-48	Sec. 5.4.5.1 p. 5-76a Fig. 5-49 through 5.54

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete mat foundation bearing on rock. 9 ft. thick with a 2 ft. thick concrete slab. Above the bottom liner plate.	sand and gravel bedrock	14-19 ft.	Bedrock 8,500 to 11,500 fps.	Depth: between 14 and 19 ft.	Not available.	Stick model with fixed base	Not available.	Not available.	Not available.
Sec. 5.2, p. 5-11a	Sec. 2.7.1 p. 2-30 Sec. 2.7.4.3 p. 2-37	Sec. 2.7.1 p. 2-30 Sec. 2.7.4.3 p- 2-37	Sec. 2.7.3.4 p. 2-34	Sec. 2.7.4.3 p. 2-37		Fig. 5-47			

SEISMIC REVIEW TABLE

STRUCTURES		
	DESIGN CRITERIA	
DAMPING OBE/SSE (% critical damping)	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Reactor Building: 2.0/2.0 Concrete Equipment Supports: 2.0/3.0 Steel Framed Structure: a) Bolted or riveted 2.5/2.5 b) Welded 1.0/1.0 Prestressed concrete structures 2.0/5.0	a) C = (1.0 ± 0.05) D + 1.5P + 1.0T b) C = (1.0 ± 0.05) D + 1.25P + 1.0T′ + 1.25E c) C = (1.0 ± 0.05) D + 1.0P + 1.0T + 1.0E′ d) C = (1.0 ± 0.05) D + 1.0W_t + 1.0 P_t	Reactor Building: ACI 318-63 ACI 301-66 (modified) AISC Manual of Steel Construction ASME BPVC Sect. III, VIII and IX ASA N 6.2-1965
Sec. 5.2.1.2.11 p. 5-18a	Sec. 5.2.3.2 p. 5-40	Sec. 5.2.3.1, p. 5-39 Sec. 5.2.2.4.1, p. 5-31

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE (% critical damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA		
			LOAD COMBINATION		ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Vital Piping	0.5/0.5	Analytical and Testing	Design loads + DBE loads	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$	ASME BPVC Sec. III USAS B31.1.0 USAS B31.7
Welded Steel Plate Assemblies	1.0/1.0		Design loads + SSE loads	$P_m + P_b \leq 1.2(1.5\ S_m)$	
			Design loads + SSE loads + Pipe rupture	$P_m \leq 2/3\ S_\mu$ $P_L + P_b \leq 2/3\ S_\mu$	
Sec. 5.2.1.2.11, p. 5-18a		p. 5-10 p. 5-76b	Sec. 4.1.2.5, p. 4-3		Table 4-2 , p. 4-38; and Sec. 4.1.3, p.4-5

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Not available.	Not available.	Not available.

SEISMIC REVIEW TABLE

Docket Number

50-320

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY mm	HOR. g	VERT. g					
Three Mile Island Nuclear Station Unit 2 Reactor type: PWR Containment type: 6 buttresses with shallow dome (pre-stressed concrete) NSSS manufacturer: Babcock and Wilcox Architect Engineer: Burns and Roe	0.06	0.04	VII	0.12	0.08	Golden Gate, 1957 El Centro, 1940 Synthetic time-history for floor response spectra	Vertical & Horizontal Components were considered to act simultaneously	SRSS. Closely spaced modes combined directly	Acceleration response Spectra for ½SSE were partially developed from "Golden Gate Park S.F. March 1957" Earthqk. Then it is modified in the low frequency region by the 1940 El Centro Earthquake - normalized to basic ground motion of 0.06g (OBE)	Time-History Method Using simulated ground motion.
Unit 2: 11-69/5-78	Sec 3.7.1.1 p. 3.7-1	Sec 3.7.2.8 p. 3.7-5		Sec 3.7.1.1 p. 3.7-1	Sec 3.7.2.9 p. 3.7-5	Sec 3.7.1.2 p. 3.7-1	Sec 3.7.2.9 p. 3.7-5	Sec 3.7.3.4 p. 3.7-8	p. 2.5-11 Fig. 2.5-8 Sec. 3.7.1.2 p. 3.7-1	Sec. 3.7.2.6 p. 3.7-5

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
The foundation slab is mild-steel reinforced circular mat Depth: 11 feet	The station is founded on unweathered shales and sandstones of Gettysburg Formation.	NOT AVAILABLE	NOT AVAILABLE	Water levels occurred generally at a depth in excess of 15 ft & ranged from 14 to ft. The ground water level occurred at a max. 6.2 ft above the top of rock with less than one ft of head above the soil-rock interface at one pt. of observation.	No large dams exist immediately upstream of the site.	Stick model with rock springs	NOT AVAILABLE	NOT AVAILABLE	NOT AVAILABLE
Sec. 1.2.3.1.1 p. 1.2-3	Sec 2.5.1 p. 2.5-7	2.9		Sec 2.4.13.2 p. 2.4-26	Sec 2.4.4 p. 2.4-12	Sec 3.7.1.6 p. 3.7-3,4			

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="3">STRUCTURES</th></tr>
<tr><th rowspan="2">DAMPING
OBE/SSE
(% of critical damping)</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td>Welded steel plate assemblies 1.0/1.0
Welded steel framed structures 2.0/2.0
Bolted steel framed structures(riveted) 2.5/2.5
Reinforced concrete equipment supports 2.0/3.0
Reinforced concrete frames & buildings 3.0/5.0
Prestressed concrete structures 2.0/5.0

Cable Tray Hangers (lateral direction) 5.0/10.0

Table 3.7-1
p. 3.7-13</td><td>Unit Load: A / B / C / D / E / F / G
Dead load: 0.95 / 0.90 / 0.95 / 0.87 / 0.95 / 0.95 / 0.90
Int. pres.: 1.50 / 1.25 / 1.25 / 1.00 / - / 1.00 / 1.00
Prestress: 1.0 / 1.0 / 1.0 / 1.0 / 1.0 / 1.0 / 1.0
Wind load: - / - / 1.25 / - / - / - / -
Tornado load: - / - / - / - / 1.25 / - / -
Earthquake: - / 0.81 / - / 1.0 / - / - / 0.81
Thermal norm: 1.25 / 1.25 / 1.25 / 1.25 / 1.25 / 1.25 / 1.25
Accident: 1.00 / 1.00 / 1.00 / 1.00 / - / 1.00 / 1.00
Thermal incr.

Table 3.8.-1,-2</td><td>1. ACI 318-63
ACI 318-71

2. AISC-1965

Sec. 3.8.1.2
p. 3.8-2</td></tr>
</table>

Unit Load	A	B	C	D	E	F	G
Dead load	0.95	0.90	0.95	0.87	0.95	0.95	0.90
Int. pres.	1.50	1.25	1.25	1.00	-	1.00	1.00
Prestress	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Wind load	-	-	1.25	-	-	-	-
Tornado load	-	-	-	-	1.25	-	-
Earthquake	-	0.81	-	1.0	-	-	0.81
Thermal norm	1.25	1.25	1.25	1.25	1.25	1.25	1.25
Accident	1.00	1.00	1.00	1.00	-	1.00	1.00
Thermal incr.							

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE (% of critical damping)		**METHOD OF QUALIFICATION**	**DESIGN CRITERIA** — **LOAD COMBINATION**			**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
				Components	Piping	
Steel Piping	0.5/0.5	Analytical procedure 1. Equivalent Static Load Method 2. Dynamic Analysis Method	I. Max Operating Loads + ½SSE (upset)	$P_m \leq 1.0S_m$ $P_L+P_b \leq 1.5S_m$	$P_m \leq S_m$ $P_L+P_b \leq 1.5S_m$	1. ASME, B&PV Code Section III 2. ANSI B31.7
			II. Max Operating Loads + SSE (emergency)	$P_m \leq 1.2S_m$ $P_L+P_b \leq 1.2(1.5S_m)$	$P_m \leq 1.2S_m$ or S_y $P_L \leq 1.8S_m$ or $1.5S_y$ $P_L+P_b \leq 1.8S_m$ or $1.5S_m$	
			III. Max Operating + SEE + Pipe Rupture Loads Faulted	$P_m \leq 2/3S_u$ $P_L+P_b \leq 2/3S_u$	$P_m \leq$ $P_L \leq$ $2/3S_u$ $P_L+P_b \leq$	
			P_b = Primary bending stress P_L = Primary local membrane stress P_m = Primary general membrane stress S_m = Allowable stress S_y = Minimum yield strength at temp. S_u = Ultimate strength of material at temp. For components: Table 3.6-1, p. 3.6-5 Table 5.2-4, p. 5.2-34 For piping: Table 5.2-3, p. 5.2-33			
Table 3.7-1 p. 3.7-13		Sec 3.9.1.2.1 p. 3.9-1,-2				Table 3.6-1 p. 3.6-5

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
NOT AVAILABLE	TESTING Sec 3.10.1.3 p. 3.10-2	NOT AVAILABLE	NOT AVAILABLE

SEISMIC REVIEW TABLE

Docket Number
50-344

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Trojan Nuclear Plant, Unit No. 1 Reactor type: PWR Containment type: 3 buttresses with hemispherical dome (prestressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Bechtel	0.15	0.10	VIII	0.25	0.17	Synthetic time history	Horizontal combined with vertical component combined absolutely	SRSS	Developed by Dr. I. M. Idriss for 2% critical damping. For other damping values Newmark's amplification factors were used.	For Westinghouse equipment: horizontal and vertical seismic were used. They were compared with the horizontal and vertical floor response spectra developed by Bechtel Corporation. Time-history used to generate response spectra BC-TOP-4
2-71/ 11-75	Sec. 2.5 p. 2.5-19 Sec. 3.7 p. 3.7-1	Sec. 3.7 p. 3.7-1	Sec. 2.5 p. 2.5-19	Sec. 3.7 p. 3.7-1	Sec. 3.7 p. 3.7-1	Sec. 3.7 p. 3.7-3	Sec. 3.7 p. 3.7-8 p. 3.7-12	Sec. 3.7 p. 3.7-22	Sec. 3.7 p. 3.7-2 Fig. 3.7-1 & 3.7-2	Sec. 3.7 p. 3.7-31

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
For containment: Rigid base mat foundation. Depth is not available. Administration building supported by steel H-piles which go to rock 15 ft to 53 ft below grade. Sec. 3.7, p. 3.7-9 Sec. 3.7, p. 3.7-4 TYPE (cont.) * soft clayed silt to silty clay with varying amounts of intermixed fine sand and layers of silty fine sand. Sec. 2.5, p. 2.5-9	The site is underlaid by bedrock and recent alluvium. The bedrock is volcanic in origin and consists principally of tuffs, tuff breccias, agglomerates, and basalt flow. Alluvium consists of soft to very*	The thickness of the alluvium is considered to be close to 280ft. The upper approx. 80 to 100 ft of the alluvium: soft to very soft clayed silt. At 50 ft depth range: decomposed wood fragments and vegetation**	4500 fps to 5000 fps. Sec. 2.5 p. 2.5-15 DEPTH (cont.) ** upper 25 ft to 35 ft. Predominately silty fine sand. All holes in the alluvium encountered principally soft clayed silt between 30 ft to 90 ft. Sec. 2.5, p. 2.5-9	Wells vary in depth from 50 feet to over 200 feet. Sec. 2.4 p. 2.4-54	Grand Coulee Dam at Columbia River mile 597. Sec. 2.4 p. 2.4-33	The dynamic analysis was performed using stick model with fixed-base assumption. Results were compared with respect to flexible-base model and found to be conservative. Sec. 3.7 p. 3.7-6	0.7×10^6 psi Sec. 2.5 p. 2.5-12	Not available.	Not available.

SEISMIC REVIEW TABLE

<table>
<tr><th colspan="6">STRUCTURES</th></tr>
<tr><th colspan="4" rowspan="2">DAMPING
OBE/SSE</th><th colspan="2">DESIGN CRITERIA</th></tr>
<tr><th>LOAD COMBINATION</th><th>ACCEPTANCE CRITERIA
& ALLOWABLE STRESSES</th></tr>
<tr><td></td><td colspan="3">Stress Level</td><td rowspan="3">$C=1/\phi\ \{(1.0\pm0.05)D+1.5P+1.0T_A+1.0F\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.25P+1.0T_A+1.0H_A+1.25E+1.0F\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.25P+1.0T_o+1.25H_o+1.25E+1.0F\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.0H_A+1.0R+1.0F+1.25E+1.0T_A\}$
$C=1/\phi\ \{(1.0\pm0.05)D\pm1.25H_o+1.0R+1.0F+1.25E+1.0T_o\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.0P+1.0T_A+1.0H_A+1.0E'+1.0F\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.0P+1.0T_o+1.25H_o+1.0E'+1.0F\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.0H_A+1.0R+1.0E'+1.0F+1.0T_A\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.25H_o+1.0R+1.0E'+1.0F+1.0T_o\}$
$C=1/\phi\ \{(1.0\pm0.05)D+1.0A+1.0F+1.0T_o\}$

For the combinations of category I structures other than containment refer to p. 3.8-13.</td><td rowspan="3">ACI 315-65
ACI 318-63
AISC 6th edition (1967)
ASCE paper no. 3269</td></tr>
<tr><td></td><td>Low</td><td>Working</td><td>At yield point</td></tr>
<tr><td>Steel Structure
Prestressed concrete
Reinforced concrete</td><td>1.0</td><td>2.0</td><td>5.0</td></tr>
<tr><td colspan="4">Sec. 3.7
Table 3.7-1
p. 3.7-3</td><td>Sec. 3.8
p. 3.8-38</td><td>Sec. 3.8
p. 3.8-12, 33</td></tr>
</table>

SEISMIC REVIEW TABLE

<table>
<tr><td colspan="5">MECHANICAL & PIPING</td></tr>
<tr><td rowspan="2">DAMPING OBE/SSE (% critical damping)</td><td rowspan="2">METHOD OF QUALIFICATION</td><td colspan="2">DESIGN CRITERIA</td></tr>
<tr><td>LOAD COMBINATION</td><td>ACCEPTANCE CRITERIA & ALLOWABLE STRESSES</td></tr>
<tr><td>Stress Level

Low / Working / At yield point

Vital piping: 0.5 / 0.5 / 0.5

Sec. 3.7
Table 3.7-1</td><td>Analytical and testing.</td><td>For reactor vessel internals:
Normal+OBE < ASME, BPVC Code, Sec. III for upset condition.
For ANSI B31.7 Class II and III and ANSI B31.1.0 seismic category I piping systems:
—For O.B.E.:
$S_T = S_{OBE} + S_{1p} + S_{wT} \leq 1.2S_h$
where: S_T = maximum total longitudinal stress
S_{OBE} = maximum bending stress due to O.B.E.
S_{1p} = longitudinal pressure stress
S_{wT} = bending stress due to weight effect
S_h = basic material allowable stress at maximum (hot) temperature
—For S.S.E.:
$S_{T(S.S.E.)} = S_{SSE} + S_{1p} + S_{wT} \leq 1.8S_h$
where: $S_{T(S.S.E.)}$ = maximum longitudinal stress
S_{SSE} = maximum bending stress due to SSE
Sec. 3.7; p. 3.7-12; p. 3.7-26.</td><td>For reactor vessel internals:
ASME, BPVC Code, Section III

For piping:
ANSI B31.7 and ANSI B31.1.0

Sec. 3.7; p. 3.7-12
Sec. 3.7; p. 3.7-26</td></tr>
</table>

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Analytical and Testing Sec. 3.10 p. 3.10-1	Not available. Sec. 3.10 p. 3.10-2	IEEE 344-1971 Sec. 3.10 p. 3.10-1

SEISMIC REVIEW TABLE

Docket Number
50-250,251

NAME AND NSSS TYPE OF THE PLANT	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
CP/OL ISSUE DATE	HOR. g	VERT. g	INTENSITY mm	HOR. g	VERT. g					
Turkey Point Plant Unit No. 3 & 4 Reactor type: PWR Containment type: 6 buttresses with shallow dome (pre-stressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Bechtel	0.05	0.033	VII	0.15	0.10	Synthetic time history	Vertical & Horizontal Components Applied Simultaneously	SRSS (Response Spectrum Analysis) Sec. 5.1 p 5.1.3-13 For reactor internals Summing the Absolute values obtained for all modes.	The Response Spectra used are those based on TID-7074 scaled to the appropriate ground accel. (Fig. 5A-1 & 2) Ref. Report to the AEC Regulatory Staff. Dockets No. 50-250 & 50-251 by N. M. Newmark & W. J. Hall, p. 5) Housner	TIME HISTORY METHOD
Unit 3: 4-67/7-72 Unit 4: 4-67/4-73	Sec. 2.11 p. 2.11-2	Sec. 2.11 p. 2.11-2		Sec. 2.11 p. 2.11-2	Sec. 2.11 p. 2.11-2	p. 5.1.3-11	Appen. 5A p. 5A-12	Appen. 5A p. 5A-9b	Sec. 5.1 p. 5.1.3-13	Sec. 5.1 p. 5.1.3-11 REV. 5 - 8-28-70 6 - 10-2-70

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE					% Critical Damping	
For containment: reinforced concrete slab. Thickness: 10 ½ feet Sec. 5.1 p. 5.1.2-1	Organic, Mangrove swamp soils overlies the Miami oolite bedrock formation which extends to about 20' below sea level (site elev less than 10') Small voids and solutions channels are present. - - - - - Below this are the Fort Thompson*	4ft to 8ft of swamp soils - overlies the Miami oolite bedrock formation.. * - - - - - Extends to 70ft -below sea level	NOT AVAILABLE TYPE THICKNESS (cont.) Formation (Limestone and calcareous sandstone - - - - - The Tamiami Formation (clayey and calcareous marl indurated locally to limestone with beds of silty and shell sands) and the Haw-~~thorne and~~ Tampa Formations Vol. 1, Sec. 2.9 p. 2.9-4	UNCLEAR INFORMATION Sec. 2.10 p. 2.10-1	NOT AVAILABLE	FIG. 5.1-13 indicates stick model with soil springs p. 5.1.3-13	NOT AVAILABLE	O.B.E./S.S.E. Soil: 5.0/10.0 Vol. 1 Append. 5A p. 5A-13	Composite with Soil: 5.0/7.5

SEISMIC REVIEW TABLE

STRUCTURES

DAMPING OBE/SSE	(% critical damping)	DESIGN CRITERIA: LOAD COMBINATION	DESIGN CRITERIA: ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Welded steel framed structure:	2.0/2.0	For class I structure outside the containment structure:	ACI 318-63
Bolted steel framed structure:	2.0/2.0	$Y=1/\phi(1.25D+1.25E)$ $Y=1/\phi(1.25D+1.0R)$	AISC Manual of Steel Construction (6th edition)
Concrete equipment supports on another structure:	2.0/2.0	$Y=1/\phi(1.25D+1.25H+1.25E)$ $Y=1/\phi(1.0D+1.0E')$	
Prestressed concrete containment structure:	2.0/5.0	where:	
Prestressed containment including interior concrete and soil composite:	3.5/7.5	Y = regular D yield strength of the structure. D = dead load of structure and equipment plus any other permanent loads contributing stress. In addition, a portion of "live load" is added when such load is expected to be present when the unit is operating.	Append. 5A, p. 5A-5 Sec. 5.1, p. 5.1.8-1
R.C. frames and buildings:	3.0/5.0	R = force or pressure on structure due to rupture of any one pipe. H = force on structure due to restrained thermal expansion of pipes under operating conditions.	LOAD COMBINATION (cont.)
		E = design earthquake load. E' = maximum earthquake load. W = wind load. (to replace E in the above load equation whenever it produces higher stresses than E does)	ϕ = 0.70 for tied comp. members. ϕ = 0.9 for fabricated structure of steel.
Append. 5A p. 5A-13		ϕ = 0.9 for R.C. in flexure. ϕ = 0.85 for tension, shear, bond, and anchorage in R.C. ϕ = 0.75 for spirally R.C. comp. members (cont.)	Vol. 1, Append. 5A p. 5A-5

SEISMIC REVIEW TABLE

MECHANICAL & PIPING						
DAMPING OBE/SSE (% of Critical Damping)		METHOD OF QUALIFICATION	DESIGN CRITERIA			
			LOAD COMBINATION			ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
			LOADING COMBINATIONS	VESSELS	PIPING	
Welded Steel Plate Assemblies	1.0/1.0	For Class I = Analysis and testing	Normal Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq S_m$ $P_L + P_B \leq S$	ASME BPVC Sec. III USAS B 31.1 Code for piping.
Steel Piping	0.5/0.5		Normal + Design Earthquake Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	
			Normal + Maximum Potential Earthquake Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	
			Normal + Pipe Rupture Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	
Append. 5A p. 5A-13		Vol. 1 Append. 5A p. 5A-12 p. 5A-17	Append. 5A p. 5A-6, Table 5A-1			Append. 5A, Table 5A-1 p. 5A-8

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available.	Tests and inspections. Vol. 2 Sec. 8.5 p. 8.5-1 & p. 8.5-2	"Electrical cable trays and DC battery racks are being checked for 'g' loadings obtained from the spectrum curves of the supporting floors. Motor control center and load centers have been shaker table tested to demonstrate no-loss-of-function capability under the maximum hypothetic earthquake. Mechanical and electrical equipment has been purchased under specifications that include a description of the seismic design criteria for the plant." p. 5A-16, B-37	

SEISMIC REVIEW TABLE

Docket Number
50-271

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Vermont Yankee Nuclear Power Station Reactor type: BWR Containment type: Mark I (steel) NSSS Manufacturer: General Electric Architect Engineer: Ebasco	0.07	0.046	V to low VII	0.14	0.093	1952 Taft earthquake N69°W	Each horizontal combined with the vertical simultaneously, resulting two distinct seismic cases.	SRSS	Housner spectra	Time-history method using earthquake N69°W component of Taft earthquake normalized to 0.07g (0.14g). See also "additional information concerning seismic analysis of piping" in App. I.
12-67/3-72	p. 2.5-9	p. 12.2-6		p.2.5-9	p.12.2-6	App. A	App. C, Sec. C.2.6 p.C.2-22	App. A p. A.5-6	See App. A., Sect. 5, Fig. 10	Question C-1, App. I, p. I.2-144

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Concrete mat. depth is not availalbe. All Class I structures except main stack are founded on bedrock. The main stack rests on end bearing steel piles which transfer the loads to the bedrock. Questions 12.18 12.19 12.22 App. I, p. I2-69	Glacial deposits from pleistocene age which consists of hard biotite gneiss. Rock type = Oliverian Plutonic Series Sec. 2.5.1,p. 2.5-1	30 ft. of glacial over burden above local bedrock. Sec. 2.5.1, p. 2.5-1	6,500 fps Sec. 2.5.2.5.2 p. 2.5-6	Unclear information (About @ El 230 and existing ground surface is @ 250 from boring logs presented in sec. 2.5)	1. Vernon Dam is about 3,500 ft. downstream. 2. Other dams are 32, 75 and 132 miles upstream. But have relatively low heads from 29 to 62 ft. Sec. 2.4 p. 2.4-1	Lumped mass with soil springs Fig. 3, App. A.1	1.53×10^6 lb/in^2 Sec. 2.5.2.5.2, p. 2.5-6	Not available	Not available

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE (% critical damping)	**DESIGN CRITERIA**		
	LOAD COMBINATION (Allowable Stress)		**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
1. Reinforced concrete structures 5.0 2. Steel frame structure 2.0 3. Bolted or riveted assembly 2.0	1. D + L + E	1. Normal allowable code stresses are used. No increase in design stresses for the load combinations considered is premitted.	1. ACI 318-63 2. AISC (1963) "Allowable Stress Design."
	2. D + L + R + E′ D + L + W′	2. Yield stresses for ductile materials 0.85 times of ultimate strength concrete.	
	D = Deal load , L = Live load E = OBE E′ = DBE	R = Jet force or pressure due to rupture of one pipe Sec. 12.2.1, p. 12.2-2	
	Note that no load factors were applied to the equations above because no plastic strength design for steel structures or ultimate strength design for concrete was used. "Allowable stress design."		
Sec. 12.2.1.2.1, p. 12.2-6	Question 12.15, App. I, p. I.2-66		Sec. 12.2.1, p. 12.2-1

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE (% critical damping)	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
-Welded assembly (Equipment and supports) 1.0 -Vital Piping System 0.5 Sec. 12.2.1.2.1, p. 12.2-6	1. Analytical 2. Testing App. C	Primary containment L.C. Normal & Upset 1. DL 2. Design pressure 3. Design temperature 4. Piping and mechanical loads 5. Design basis earthquake Emergency condition loads 1. Dead load 2. Design pressure 3. Design temperature 4. Piping and mechanical loads 5. Maximum hypothetical earthquake For flooded containment condition 1. Dead weight 2. Design basis earthquake 3. Flooding water load App. C. pg. C.2-30	Stress Limit ASME B&PV Code, Sect. III, Subsection B. Membrane stress intensity $S_A = 1.0\ S_M = 17{,}500$ psi Primary local membrane and bending : $S_{limit} = 1.5\ S_M = 26{,}250$ psi Membrane plus secondary bending $S_{limit} = 3.0\ S_M = 52{,}500$ psi Primary local stress = 90% of yield strength @ design temperature $S_a = 0.90 \times 33{,}700 = 30{,}330$ psi Primary local stress = 90% of yield strength @ 100°F $S_a = 0.90 \times 38{,}000 = 34{,}200$ psi

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

SEISMIC REVIEW TABLE*

Docket Number
50-29

NAME AND NSSS TYPE OF THE PLANT / CP/OL ISSUE DATE	EARTHQUAKE DATA**						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Yankee Rowe Nuclear Power Station. Reactor type: PWR Containment type: Spherical (steel) NSSS Manufacturer: Westinghouse Architect Engineer: Stone and Webster Engineer Corp. 11-57/7-60	← No Seismic Analysis Performed →		VI							

* Remarks: Information obtained from BNL Docket Search and SEPB Report by LLL "Seismic Design Bases and Criteria for Yankee Rowe Generating Station", EDAC 175-130.02, January 1979.

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
"All structures and equipment should be founded on spread footings. Where there is possibility of heaving due to frost action, footings should be carried to a minimum depth of 5'-0" below ground surface"- summary of Stone and Webster's structural design requirements 10-17-57. General design of turbine generator foundation conform to "GET-1749A"- "turbine generator foundations" and Stone and Webster Reinforcing Standard for Turbine Supports ' 4-20-48	"The plant is situated on medium to fine sands with some clay and silt, cobbles and boulders".	Not available	Not available	Not available	Sherman Dam	No soil-structure Interaction analysis			

SEISMIC REVIEW TABLE

STRUCTURES		
DAMPING OBE/SSE	DESIGN CRITERIA	
	LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
None used	"Neither structures nor equipment were classified into seismic categories, e.g., seismic category I or equivalent, but instead were classified as safety related or non-safety related. These systems were designed and analyzed in accordance with the design codes in effect in 1955. For structures, the design of lateral load restraint systems was dictated by wind requirements. No lateral force provisions were made for internal structures or equipment."	AISC American Standard Building Code requirements A58.1-1955 ACI 318-56 ASTM - specifications for structural steel for bridges. ASA A56.1 - 1952 Stone and Webster "Summary of Structural Design Requirements, Yankee Atomic Electric Co." J. O. No. 9699, October 1957.

SEISMIC REVIEW TABLE

MECHANICAL & PIPING			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
None used	None used	Not available	ASME B and PV Code, Section VIII "Unfired Pressure Vessels" 1955 and code case 1226 ASTM specification for A300 (Class A201, Grade B, Firebox Quality)

SEISMIC REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
None used	None	"Electrical penetrations, control room systems, etc, were designed based on nuclear, mechanical and functional criteria. No provisions for lateral loads."	Not available

SEISMIC REVIEW TABLE

Docket Number
50 - 295, 304

NAME AND NSSS TYPE OF THE PLANT CP/OL ISSUE DATE	EARTHQUAKE DATA						METHOD OF COMBINATION		DESIGN SPECTRA	
	OBE		SSE			EARTHQUAKE TIME HISTORY	NO. OF EARTH. COMP. USED AND ITS COMB.	MODAL COMB.	TYPE OF GROUND DESIGN SPECTRA	METHOD OF GENERATION OF FLOOR RESPONSE SPECTRA
	HOR. g	VERT. g	INTENSITY MM	HOR. g	VERT. g					
Zion Nuclear Plant Unit 1 and 2 Reactor type: PWR Containment type: 6 buttresses with shallow dome (pre-stressed concrete) NSSS Manufacturer: Westinghouse Architect Engineer: Sargent and Lundy Engineers Unit 1: 12-68/4-73 Unit 2: 12-68/11-73	0.08 p.2.11-2	0.05 p.2.11-2	VII Q.2.26-1	0.17 p.2.11-3	0.11 p.2.11-3	Compared with the 1940 El Centro (N-S) earthquake record with maximum acceleration of 0.08g. Amend. 18 Q.5.79	Each horizontal was combined with the vertical components simultaneously. Amend. 14 Q.4.23	SRSS with closely spaced modes combined by absolute sum method (response spectrum) Amend. 14 Q.4.23	Design response spectra using 1940 El Centro (N-S) earthquake record with maximum acceleration of 0.08g at the rock level. Amend. 19 Q.5.83	Time-history method using 1940 El Centro (N-S) earthquake record. Amend. 14, Q. 4.25 Amend. 19, Q. 5.83

SEISMIC REVIEW TABLE

FOUNDATION AND LIQUEFACTION ASSESSMENT						SOIL - STRUCTURE INTERACTION			
TYPE OF FOUNDATION AND ITS DEPTH	BEARING INFORMATION			GROUND WATER TABLE	DAM	METHOD OF MODELLING	G_s PROFILE	MATERIAL DAMPING OF SOIL	LIMITATION ON MODAL DAMPING
	TYPE	THICKNESS	V_s PROFILE						
Reinforced concrete slab 9ft thick p. 5.1-5	The plant will be founded on relatively firm partly preconsolidated. Pleistocene glacial deposits. Formations below the site consist of: 1) 24-33 ft. of lake deposits-sand, gravel and pockets of peat and organic material. 2) Glacial deposits extending to a depth of 102-116 ft. below the surface - silt, clay, sand and gravel.		Not available 3) Niagare dolomite is 250' thick 4) Lower bedrick formations consists of sandstone and dolomite, some shale and silt-stone layers. Several thousands of ft. thick. 5) Precambrain basement.*	Ground water is near the surface over much of the site area p. 2.9-5	Not available	Aux. building was modelled as fixed base assumptions with lumped mass building model. Reactor building model has a rocking soil spring only. A comparison study was made with a soil model by finite element mesh. Amend. 18 Q. 5.79 Amend. 14 Q.5.3,Q.4.23	Not available	Soil % critical damping: OBE 2 DBE 5 Q. 5.80	Not available

p. 2.9-4

* Type and thickness of bearing information are presented together.

SEISMIC REVIEW TABLE

STRUCTURES			
DAMPING OBE/SSE (% critical damping)		**DESIGN CRITERIA**	
		LOAD COMBINATION	**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Reactor containment:	0.5/2.0	1) $C = (1/\phi)\ (1.05\ D + 1.25\ P + 1.0\ T + 1.25\ E)$ 2) $C = (1/\phi)\ (1.05\ D + 1.5\ P + 1.0\ T)$ 3) $C = (1/\phi)\ (1.05\ D + 1.0\ P + 1.0\ T + E')$ C = Required yield strenght of the structure as defined below D = Dead loads P = Design accident pressure T = Thermal loads due to the temperature gradient through the wall and expansion of the liner and based on a temperature corresponding to the factored design accident pressure E = Operating basis earthquake (OBE) load E′ = Design basis earthquake (DBE) load W = Wind load ϕ = Capacity reduction factor	ACI Code 318-63 refer to page 5.1-41 for ϕ values. AISC Manual of Steel Construction (6th Edition)
Q. 4.23		p. 5.1-38	p. 5.1-41

SEISMIC REVIEW TABLE

MECHANICAL & PIPING					
DAMPING OBE/SSE (% critical damping)	**METHOD OF QUALIFICATION**	**DESIGN CRITERIA** — **LOAD COMBINATION**	Pressure Vessels	Pressure Piping	**ACCEPTANCE CRITERIA & ALLOWABLE STRESSES**
Piping OBE = 0.5	Analytical and Testing	1) Normal condition	a) $P_m \leq S_m$ b) P_m(or P_L) + $P_B \leq 1.5\ S_m$ c) P_m(or P_L) + P_B + Q ≤ 3.0 S_m	a) $P_m \leq S$ b) P_m(or P_L) + $P_B \leq S$	ASME B&PV Code Section III, Nuclear Vessels for limit curves: WCAP 5890, Rev. 1
		2) Upset condition	a) $P_m \leq S_m$ b) P_m(or P_L) + $P_B \leq 1.5\ S_m$ c) P_m(or P_L) + P_B + Q ≤ 3.0 S_m	a) $P_m \leq 1.2$ S b) P_m(or P_L) + $P_B \leq 1.2$ S	
		3) Emergency condition	a) $P_m \leq 1.2\ S_m$ or S_y whichever is larger b) P_m(or P_L) + $P_B \leq 1.5$ (1.2 S_m) or 1.5 S_y whichever is larger	a) $P_m \leq 1.2$ S b) P_m(or P_L) + $P_B \leq 1.5$ (1.2 S)	
		4) Faulted condition	Design limit curves as discussed in the text	Design limit curves as discussed in the text	
P. Q. $.32-1	Appendix D Amend. 14 Q. 4.23 p. Q4.23-3				Appendix D

P_m = Primary general membrane stress intensity

P_L = Primary local membrane stress intensity

P_B = Primary bending stress intensity

Q = Secondary stress intensity

S_m = Stress intensity from ASME B&PV Code, Section III, nuclear vessels

S_y = Minimum specified material yield (ASME B&PV Code, Section III, Table N-421 or equivalent)

S = Allowable stress from USASI B31.1 Code for pressure piping. Table B1-2, Appendix D

SEISMIC-REVIEW TABLE

ELECTRICAL EQUIPMENT			
DAMPING OBE/SSE	METHOD OF QUALIFICATION	DESIGN CRITERIA	
		LOAD COMBINATION	ACCEPTANCE CRITERIA & ALLOWABLE STRESSES
Not available	Not available	Not available	Not available

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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7. AUTHOR(S)
M. Subudhi, J. Lane, M. Reich, B. Koplik

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9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Brookhaven National Lab
Upton, N.Y. 11973

DATE REPORT ISSUED
MONTH May | YEAR 1980

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U.S. Nuclear Regulatory Commission
Division of Operating Reactors
Seismic Review Group
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

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FIN No. A3326

13. TYPE OF REPORT
Final

PERIOD COVERED (Inclusive dates)
October 1979-April 1980

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)
The Seismic Review Table is a summary of Engineering Design parameters that were employed in the seismic analysis and design of nuclear power plants. The table covers 71 reactors licensed to operate by the U.S.N.R.C. The information contained is listed plant by plant and consists of OBE and SSE "g" level and Modified Mercalli Intesity; Earthquake Time History used to develop the ground response spectra or as input in the dynamic analysis; Number of Earthquake Components used and Method of Combining Them; Method of Modal Combination; Type of Ground Design Spectra; Method of Generation of Floor Response Spectra; Type of Foundation and Depth; Type, Thickness, Shear Wave Velocity and Shear Modulus Profile of the Surrounding Sub-grade Soil and Bedrock; Ground Water Table Depth; nearby Dams; Modelling Method used for soil-structure interaction; Material Damping of Soil; Limitation on Modal Damping. Damping Values and Loading Combinations, and Acceptance Criteria for Category I Structures, Mechanical Equipment, Piping, and Electrical systems. The goal of the Seismic Review Table is to provide a reference of the available information relevant to the seismic design of currently licensed nuclear power plants.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

seismic data, earthquake design, dynamic analysis, soil-structure interaction
load combinations, design criteria

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
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NRC FORM 335 (7-77)

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Friday, April 01, 2011 3:24 PM
To: Lee, Richard
Subject: conference phone call at 4:30 EDT

Richard I think you said the phone call today was at 4:30 EDT. Dana

4/307

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Friday, April 01, 2011 4:36 PM
To: Lee, Richard
Subject: call today

I have received nothing either.

4/308

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Friday, April 01, 2011 7:03 PM
To: Adams, Ian
Cc: Brinkman, Bill; Narendra, Blake; Hurlbut, Brandon; Sheron, Brian; Butnitz, Bob (pacbell.net); Smith, Haley; McFarlane, Harold; Kelly, John E (NE); Grossenbacher, John (INL); Pitzer, Karrie S.; Chambers, Megan (S4); Owens, Missy; Miller, Neile; Fitzgerald, Paige; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Steve Fetter; Aoki, Steven; Binkley, Steve; Mustin, Tracy
Subject: Shippable tanks are tiny compared with the need.

A 16,000 gallon tank is bout 60,000 liters, or 60 tons of water.

The torus of Unit 1 holds 5000 tons of water. Need 80 tanks.

Dick Garwin

4/309

Public

Bonaccorso, Amy

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr. [backflow.prevention@verizon.net]
Sent: Saturday, April 02, 2011 4:09 PM
To: NRC Allegation
Subject: Fw: EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container.

FYI NRC.

-------*Original Message*-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 4/2/2011 11:59:55 AM
To: cgjak@alaska.com; info@cgjapanatlanta.org; infocul@cgjbos.org; ryoji@japancc.org; jic@japancc.org; cgjd-pr@quest.net; japaninfo@ggjsf.org; info@cgjapansea.org; jet@ws.mofa.go.jp
Subject: Fw: EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container.

FYI. URGENT FOR JAPAN !

-------*Original Message*-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 4/2/2011 11:21:26 AM
To: POTUS Office Of The President; rick.scott@eog.myflorida.com; Senator Bill Nelson; Senator John McCain; Rep. Paul; APFN; APFN-1; mailtothechief@cnn.com; Dlind49@aol.com; leurenmoret@yahoo.com
Cc: Gordhan N. Patel; Albrodsky@aol.com; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov
Subject: EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container.

4/310

Ladies and Gentlemen:

Storage of spent rods deep in the Salt Water Ocean some 3,000 feet might find the temperature staying at 40 degrees F continuous. Same of your refrigerator at home perhaps. I have been provided with 8 NEW RADStickers from the Inventor Gordhan Patel at J.P. Labs for use by any of your personnel who will be working at the plant and these stickers attach to ID's or Drivers Licenses so the radiation levels can be measured daily, weekly and monthly. Would also recommend placing some eventually on ALL equipment then a log started to see how much radiation the equipment plus personnel are subjected to in the line of work completed at the sites now or in the future. RADTRIAGE units work inside or outside as suggested but the RADSticker can be read after each shift if necessary for workers.

Remember STORAGE OF RODS SPENT OR REMOVED FROM REACTORS means WATER WITH DRY ICE COOLING FOR TRIP TO SEA THEN DROPPING RODS going to 12,000 feet depth in THE BATHYPELAGIC ZONE below the 3,000 feet to say 12,000 FEET might find 36 degrees F. Each is suitable perhaps for storage of spent fuel rods in very deep sections of sea water. Such no doubt was the bulk of laughter when it was recommended that spent rods be deep six stored rather than underground in areas like the scrubbed YUCCA MOUNTAIN storage which might now be for members of Congress and their families to live in any Nuclear Emergency in the USA. Wonder if SUPER THICK MILL PLASTIC SHEATING WOULD ALLOW SEALING FROM SALT WATER YET LET COLD GO THROUGH TO KEEP RODS SAFE ? Check with NRC personnel on this NEW STORAGE METHOD " DEEP SIX " IDEA !

As a State of Florida Certified Plumbing Contractor let me tell you that SOLAR ENERGY/DIESEL GENERATORS should be the backup system for batteries and generators running pumps for cooling near ALL NUCLEAR POWER PLANTS AND EARTHQUAKE PROOF BLADDERS AND TANKS SHOULD HOUSE EXTRA FRESH WATER NEAR NUCLEAR REACTORS for just such an emergency. DRY ICE CAN BE DROPPED INTO THE POOLS AS WELL AS LOX OR LIQUID NITROGEN PUSHED THROUGH STEEL PIPING SUBMERGED AND EXPOSED IN THE WATER TO ALLOW COOLING BUT HOSPITALS HAVE LARGE TANKS WHICH CAN BE RELOADED WHEN RUNNING OUT LIKE I FOUND AT TAMPA GENERAL HOSPITAL when working there for 8.5 years.

Why CNN does not show super tankers pulled or pushed loaded with fresh water from even CHINA if necessary into the harbor near the stricken plants or cities to supply fresh potable water for drinking and bathing is still a mystery. 100 foot coils of black plastic piping coiled and attached to tanks with shower handles and ball valve would allow any WORKERS to be decontaminated with fresh warm water or showers but I still like the DECONTAMINATION PROCEDURES WRITTEN BY DR. DOUG ROKKE FOR THE U.S. MILITARY.

Roads should have 100 tankers with fresh water coming into the area like milk trucks to let the people fill gallon zip lock bags which can be doubling as latrine items storing urine and fecal matter to be picked up and burned. COLD WEATHER SIGNALS USE OF DRUM LINERS TWO PER PERSON STEP INSIDE THEN PULL ONE OVER YOUR HEAD AND SLEEP WARM ..... CAMP JAPAN CAN MAKE IT !

Don't start on not having fuel when there are thousands upon thousands of vehicles with gasoline tanks still containing gas sitting everywhere plus trucks and boats using diesel would power super generators which have not arrived for reasons unknown. SNAP LIGHTS or shake and snap which last 12 hours giving off light could be used by people inside rolling latrines over 55 gallon drums with seal top lids LEFT OPEN for pickup on barges and dropping or washing contents off barges at sea to make fish food. People cringed when I recommended taking the dead to barges, having a prayer service then taking barges to sea, chopping up the dead with SUPER STRONG WOOD CHIPPERS then washing down the barges after all are made into FISH FOOD. What are you going to do ? Burn the bodies releasing RADIOACTIVE PARTICLES INTO THE JAPANESE AIR, WATER, SAND AND SOIL OR ON PLANTS AS IT FALLS BACK TO THE GROUND WHILE BODIES ARE CREMATED ? Now that is smart !

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR

CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, THEN PLANT OR POUND THICK STEEL PLATES INTO THE GROUND TO ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS PUSHING THE PLATE OVER THE PATCH OF RUBBER ONLY TO STOP THE LEAK BEFORE CRACKING MORE THEN SEAL ALL IN HYDRAULIC CEMENT ! How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !

Think of Japan building a SEAWALL when they pound special plates in the ground then sealing same they POUR CEMENT FOOTERS after sealing the sea water away with pumps DE WATERING the area for a few hours at least. HOW HARD CAN THAT BE TO VISUALIZE.

Now think of 10 mil thick 3 FT SQUARE BAGS then find barges of Concrete in 80 pound bags, Take a few tanker trucks with cement trucks and load the cement trucks with the proper mixture and you have a CEMENT PLANT. Due to the radiation the CEMENT might have to come from inland in those trucks but CEMENT PUMPS like HI-RISE BUILDERS use pouring flooring UP will push the slurry mix any height required and it is CRITICAL to verify FRESH WATER and proper mixture of HYDRAULIC CEMENT to make it set up quickly.

Professionally submitted FREE to the NRC and people of Japan.

Ralph Charles Whitley, Sr. CFC0326321
Tampa Phone : (813-286-2333) SKYPE: ralphwhitleysr
040211 @ 11:21 AM Eastern Saturday

6

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SCRIBID ID: ralphwhitleysr
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SCRIBD WEB PAGE: http://www.scribd.com/ralphwhitleysr

(backflow.prevention@verizon.net) 6

Bonaccorso, Amy

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr. [backflow.prevention@verizon.net]
Sent: Sunday, April 03, 2011 12:06 PM
To: cgjak@alaska.com; info@cgjapanatlanta.org; infocul@cgjbos.org; ryoji@japancc.org; jic@japancc.org; cgjd-pr@quest.net; japaninfo@ggjsf.org; info@cgjapansea.org; jet@ws.mofa.go.jp; APFN; APFN-1; tilo@socom.mil
Cc: POTUS_Office Of The President; mailtothechief@cnn.com; Senator Bill Nelson; Rep. Paul; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov; rick.scott@eog.myflorida.com; NRC Allegation
Subject: NEW EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container THEN DAMAGE CONTROL TECHNIQUE FOLLOWED SEALING THE ...

Importance: High

LADIES AND GENTLEMEN:

NOT TOO MANY OF YOU NUCLEAR POWER PLANT OPERATORS OR OWNERS HAVE WORKED WITH FLOWING WATER FROM CRACKS IN EARTH/STEEL BULKHEADS OR EVEN POOLS OR CONCRETE ENCLOSURES UNDERGROUND.

Make no mistake, AMERICA and other Nations are at risk should US PROFESSIONALS NOT PROVIDE SOUND VERIFIABLE METHODS TO SOLVE ODD PROBLEMS YOUR NUCLEAR POWER PEOPLE DO NOT ENCOUNTER. Check with any City, State or County SEWER OR WATER DEPARTMENT PROFESSIONALS and they will confirm the information shown below to be sound and will work every single time. Key is only provide enough pressure to STOP THE LEAK not cause further failure of damaged concrete sections but allow extra rubberized seals like one would use on top of canning jars except these will be 1 inch thick but compressible rubber to seal the leak until a full pour can be accomplished with HYDRAULIC CEMENT not average cement.

FLOWING WATER CANNOT BE STOPPED WITH CEMENT POUR EVEN IN THE CONCRETE VAULTS PRESSURE WOULD PUSH IT OUT AND YOU CAN JACKHAMMER THE CEMENT WHICH WILL LIFT AWAY FROM THE FLOW OF CONCRETE THEN PREPARE THE AREA PROPERLY AND APPLY THE PILLOW OF RUBBERIZED MATERIAL THEN PLATE OF STEEL THEN PRESS FROM THE OTHER SIDE AGAINST THE PLATE WITH ITEMS YOU HAVE SEEN NAVAL PERSONNEL FIGHT WATER LEAKING THROUGH HULLS ON SUBMARINES TO SHIPS OF THE FLEET. ASK THE NAVY PERSONNEL WITH DAMAGE CONTROL HOW TO STOP THE LEAK AND THEN POUR HYDRAULIC CEMENT TO STOP ANY FURTHER EXPANSION OF THE CRACK. THEY ARE THE EXPERTS MR. PRESIDENT AND JAPANESE EMBASSY STAFF. USE THEM AND THEIR KNOWLEDGE PLUS THEY MAY EVEN LOAN YOU THE EQUIPMENT SO YOU CAN WATCH A FLICK AND DO IT EVERY TIME !

4/3/11

U.S. NAVY HAS DONE THIS IN DAMAGE CONTROL SCHOOLS AND SHIPBOARD FOR OVER 100 YEARS.

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

[[[REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, NO STEEL AS YOU HAVE ANOTHER SIDE WHERE PLATE CAN BE ATTACHED OR HELD IN PLACE WHILE WORKERS ATTACH SPECIAL SCREW BOLTS TO PRESS AGAINST THE PLATE AND RUBBER TO SEAL THE LEAK WITHOUT CAUSING

FRACTURE OF MORE. ONCE WATER IS STOPPED THEN AND ONLY THEN FILL THE ENTIRE AREA WITH HYDRAULIC CEMENT IN THAT CONCRETE UNDERGROUND FOUR WALLED CONTAINER USING ONLY FRESH WATER AFTER PUMPING DOWN ALL WATER REMAINING IN THE CEMENT PIT USED FOR ELECTRICAL CONNECTIONS AND FEEDING THE WATER TO THE SEA.

REMEMBER ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS EVEN SCREW JACKS ONLY PUSHING THE PLATE OVER THE PATCH OF RUBBER ENOUGH TO ONLY STOP THE LEAK BEFORE CAUSING MORE PRESSURE CRACKING MORE OF DAMAGED CONCRETE THEN SEAL ALL IN HYDRAULIC CEMENT !

How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !]]]

Professionally submitted FREE to the NRC and people of Japan Sunday April 3, 2011 as I fear not giving you the information could risk MASSIVE NUCLEAR EXPLOSIONS AT THE PLANT AND ENDANGER AMERICA AS WELL AS THE ENTIRE PLANET.

Have passport and medications for four weeks Mr. President and Japanese Embassy Staff should I be needed there. Japan Airlines can bring me to your Nation if necessary as this inability to follow professional recommendations is unbelievable. WHERE IS THE MAN TO HOLD HIS HAND OVER THE WATER FLOW LIKE PUTTING A FINGER IN A RESERVOIR LEAK ? IS THAT PROFESSIONALISM DROPPING CONCRETE ONTO A FIRE HYDRANT FLOW OF WATER IN A CONCRETE BOX IN THE GROUND ? 1 POUND OF WATER PRESSURE PER HOW MANY INCHES COLUMN OF WATER STORED ? I WAS TAUGHT 14.7 INCHES COLUMNAR EQUALS 1 POUND. How tall is the building and how many pounds of pressure will be pushing water through the crack. THIS IS NOT A FISH TANK FOLKS ! CHECK WITH THE NAVAL SUBMARINE SERVICE AND SHIPS OF THE NAVY....JAPANESE NAVY GOES THROUGH SIMILAR TRAINING BUT YOU HAVE A BASE IN JAPAN PLUS FLEET OFFSHORE TO DRAW INFORMATION FROM..... BORROW THEIR EQUIPMENT AND SCREW JACKS FOR THAT CEMENT SECTION.

Ralph Charles Whitley, Sr. CFC032631
Backflow Prevention, Inc.
Phone: 813-286-2333
Tampa

040311 @ 12:05 PM Eastern Sunday

Remember:

Storage of spent rods deep in the Salt Water Ocean some 3,000 feet might find the temperature staying at 40 degrees F continuous. Same of your refrigerator at home perhaps. I have been provided with 8 NEW RADStickers from the Inventor Gordhan Patel at J.P. Labs for use by any of your personnel who will be working at the plant and these stickers attach to ID's or Drivers Licenses so the radiation levels can be measured daily, weekly and monthly. Would also recommend placing some eventually on ALL equipment then a log started to see how much radiation the equipment plus personnel are subjected to in the line of work completed at the sites now or in the future. RADTRIAGE units work inside or outside as suggested but the RADSticker can be read after each shift if necessary for workers.

Remember STORAGE OF RODS SPENT OR REMOVED FROM REACTORS means WATER WITH DRY ICE COOLING FOR TRIP TO SEA THEN DROPPING RODS going to 12,000 feet depth in THE BATHYPELAGIC ZONE below the 3,000 feet to say 12,000 FEET might find 36 degrees F. Each is suitable perhaps for storage of spent fuel rods in very deep sections of sea water. Such no doubt was the bulk of laughter when it was recommended that spent rods be deep six stored rather than underground in areas like the scrubbed YUCCA MOUNTAIN storage which might now be for members of Congress and their families to live in any Nuclear Emergency in the USA. Wonder if SUPER THICK MILL PLASTIC SHEATING WOULD ALLOW SEALING FROM SALT WATER YET LET COLD GO THROUGH TO KEEP RODS SAFE ? Check with NRC personnel on this NEW STORAGE METHOD " DEEP SIX " IDEA !

As a State of Florida Certified Plumbing Contractor let me tell you that SOLAR ENERGY/DIESEL GENERATORS should be the backup system for batteries and generators running pumps for cooling near ALL NUCLEAR POWER PLANTS AND EARTHQUAKE PROOF BLADDERS AND TANKS SHOULD HOUSE EXTRA FRESH WATER NEAR NUCLEAR REACTORS for just such an emergency. DRY ICE CAN BE DROPPED INTO THE POOLS AS WELL AS LOX OR LIQUID NITROGEN PUSHED THROUGH STEEL PIPING SUBMERGED AND EXPOSED IN THE WATER TO ALLOW COOLING BUT HOSPITALS HAVE LARGE TANKS WHICH CAN BE RELOADED WHEN RUNNING OUT LIKE I FOUND AT TAMPA GENERAL HOSPITAL when working there for 8.5 years.

Why CNN does not show super tankers pulled or pushed loaded with fresh water from even CHINA if necessary into the harbor near the stricken plants or cities to supply fresh potable water for drinking and bathing is still a mystery. 100 foot coils of black plastic piping coiled and attached to tanks with shower handles and ball valve would allow any WORKERS to be decontaminated with fresh warm water or showers but I still like the DECONTAMINATION PROCEDURES WRITTEN BY DR. DOUG ROKKE FOR THE U.S. MILITARY.

Roads should have 100 tankers with fresh water coming into the area like milk trucks to let the people fill gallon zip lock bags which can be doubling as latrine items storing urine and fecal matter to be picked up and burned. COLD WEATHER SIGNALS USE OF DRUM LINERS TWO PER PERSON STEP INSIDE THEN PULL ONE OVER YOUR HEAD AND SLEEP WARM ..... CAMP JAPAN CAN MAKE IT !

Don't start on not having fuel when there are thousands upon thousands of vehicles with

gasoline tanks still containing gas sitting everywhere plus trucks and boats using diesel would power super generators which have not arrived for reasons unknown. SNAP LIGHTS or shake and snap which last 12 hours giving off light could be used by people inside rolling latrines over 55 gallon drums with seal top lids LEFT OPEN for pickup on barges and dropping or washing contents off barges at sea to make fish food. People cringed when I recommended taking the dead to barges, having a prayer service then taking barges to sea, chopping up the dead with SUPER STRONG WOOD CHIPPERS then washing down the barges after all are made into FISH FOOD. What are you going to do ? Burn the bodies releasing RADIOACTIVE PARTICLES INTO THE JAPANESE AIR, WATER, SAND AND SOIL OR ON PLANTS AS IT FALLS BACK TO THE GROUND WHILE BODIES ARE CREMATED ? Now that is smart !

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, THEN PLANT OR POUND THICK STEEL PLATES INTO THE GROUND TO ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS PUSHING THE PLATE OVER THE PATCH OF RUBBER ONLY TO STOP THE LEAK BEFORE CRACKING MORE THEN SEAL ALL IN HYDRAULIC CEMENT ! How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !

Think of Japan building a SEAWALL when they pound special plates in the ground then sealing same they POUR CEMENT FOOTERS after sealing the sea water away with pumps DE WATERING the area for a few hours at least. HOW HARD CAN THAT BE TO VISUALIZE.

Now think of 10 mil thick 3 FT SQUARE BAGS then find barges of Concrete in 80 pound bags, Take a few tanker trucks with cement trucks and load the cement trucks with the proper mixture and you have a CEMENT PLANT. Due to the radiation the CEMENT might have to come from inland in those trucks but CEMENT PUMPS like HI-RISE BUILDERS use pouring flooring UP will push the slurry mix any height required and it is CRITICAL to verify FRESH WATER and proper mixture of HYDRAULIC CEMENT to make it set up quickly.

Professionally submitted FREE to the NRC and people of Japan.

Ralph Charles Whitley, Sr. CFC0326321
Tampa Phone : 813-286-2333 SKYPE: ralphwhitleysr
040211 @ 11:21 AM Eastern Saturday

b6

Ralph Charles Whitley, Sr. CFC032631
Backflow Prevention, Inc.
4532 W. Kennedy Blvd. PMB-276
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Phone: 813-286-2333

b6

SCRIBID ID: ralphwhitleysr
SKYPE ID: ralphwhitleysr

SCRIBD WEB PAGE: http://www.scribd.com/ralphwhitleysr

backflow.prevention@verizon.net 6

Rel

From:	Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
To:	Bonaccorso, Amy; NRC Allegation; Senator Bill Nelson; rick.scott@eog.myflorida.com; mailtothechief@cnn.com; cjak@alaska.com; info@cgjapanatlanta.org; info@who.int; Inquiries; ryoji@japancc.org; Dlind49@aol.com; leurenmoret@yahoo.com
Cc:	tobin.jennifer@nrc.gov; deavers.ron@nrc.gov; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov
Subject:	Fw: ADDITIONAL INFORMATION Re: NEW EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container THEN DAMAGE CONTROL ...
Date:	Sunday, April 03, 2011 9:57:10 PM
Attachments:	SENDER_EMAILbackflow@@prevention@verizon@@net.png image0021.png
Importance:	High

Amy Bonaccorso and others:

The problem in Japan can go INTERNATIONAL in a flash without people contacting professionals as indicated below. Thank you for your kind response to the OLD MAN WHO APPEARS TO BE EXPRESSING A "RANT" when in actual Truth the man is a Professional Injured from Cancer, recovering from Surgery with a large SKIN GRAFT DEEP ON RIGHT FOREARM, Scar behind Left Ear and bleeding yet advising Dr. Doug Rokke and Dr. Leuren Moret plus the President that I would GLADLY at 70 years old replace a young worker inside that facility.

SEE YOUR RESPONSE TO THOSE WORKING FOR NRC THEN REVIEW WHAT TRAINING I HAD AND HOW A DECORATED AMERICAN VETERAN AND PLANKOWNER FOR THE U.S.S. ENTERPRISE CVAN-65 COULD POSSIBLY KNOW ANYTHING ABOUT NUCLEAR RADIATION AND ACCIDENTS OR HOW TO STOP THE RADIATION LEAKS WHICH ANY STATE OF FLORIDA CERTIFIED PLUMBING CONTRACTOR WOULD KNOW!

I will, for information, attach your message BELOW FOR REVIEW. Right now this information is SUPER CRITICAL as all of the PLanet could suffer because someone throws away the messages as from a man with a RANT ? I even gave ALL ACROSS THE PLANET THE OPEN SOLUTION TO SELECT A SUPER DEEP ABYSS AND STORE THE SPENT FUEL FROM ALL NATIONS IN A CONTROLLED TEMPERATURE...even if you loose the opportunity to make ballast or Depleted Uranium Munitions from same. Remember NRC that YUCCA MOUNTAIN SPENT BILLIONS and has been closed down for storage. Where YOU will store all of the contaminated SAND FROM IRAQ, AFGHANISTAN, PAKISTAN and perhaps LIBYA is a mystery.

I say DEEP SIX IT....12,000 feet abyss is my first choice. You do not know what it means to offer YOUR LIFE to a young person so they can survive!

I remember a PO2 Monsoor who was with EOD Unit 2 who threw himself on a grenade at a young age to save his team. DID HE RANT TOO ? No Dragon Skin Body Armor but instead wore plate vests from Interceptor. Pinnacle Armor makes Dragon Skin Body Armor and they have never had a penetration from IED or bullet fired. Gee....what is the problem there... General's have stock in Interceptor ? I carry and have carried a ballistic clipboard since 1960's in law enforcement. Latest LEVEL III clipboards were donated to Tampa Police and Hillsborough County Sheriff's Office Tactical Response Team or Special Weapons And Tactics team members. Stupid RANT again ? Imagine every military person with such protection which would save lives. My Clipboards are about $50 to manufacture and the hardest part is sealing and painting the FIBERGLASS 1/2 INCH THICK sections. BLAME UNCLE AND LAW ENFORCEMENT as they trained me along with the Shipyard when building U.S.S. Enterprise CVAN-65 as I am a PLANK OWNER !

4/3/12

DO READ AND REVIEW THE RANT BELOW AND THEN GET THOSE SPECIALISTS USING PROPER TECHNIQUES SINCE THEY HAVE LANDED IN TOKYO ! I KNOW YOU DO WANT TO STOP THE RADIATION WATER GOING INTO THE WATER AROUND THE PLANT. IF YOU GO THERE WEAR SPECIAL NAIL PROOF BOOTS !

Ralph Charles Whitley, Sr.
040311 @ 9:55 PM Eastern Tampa

-------*Original Message*-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 4/3/2011 1:10:16 PM
To: news@worldnetdaily.com; Reporter Bob Unruh WND; WorldNetDaily
Subject: Fw: ADDITIONAL INFORMATION Re: NEW EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container THEN DAMAGE CONTROL TECHNIQUE FOLLOWED SEALING THE LEAK BEFORE POURING CONCRETE OR HYDRAULIC CEMENT..

Even the Japanese NAVY know how to solve the water leaking through that crack in electrical vault piping.

Ralph

-------*Original Message*-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 4/3/2011 12:33:25 PM
To: cgjak@alaska.com; info@cgjapanatlanta.org; infocul@cgjbos.org; ryoji@japancc.org; jic@japancc.org; cgjd-pr@quest.net; japaninfo@ggjsf.org; info@cgjapansea.org; jet@ws.mofa.go.jp; APFN; APFN-1; tilo@socom.mil
Cc: POTUS_Office Of The President; mailtothechief@cnn.com; Senator Bill Nelson; Rep. Paul; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov; rick.scott@eog.myflorida.com; NRC Allegation
Subject: ADDITIONAL INFORMATION Re: NEW EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container THEN DAMAGE CONTROL TECHNIQUE FOLLOWED SEALING THE LEAK BEFORE POURING CONCRETE OR HYDRAULIC CEMENT..

REMEMBER:

ONE POUND PER SQUARE INCH EQUALS 2.3 FEET HIGH COLUMN OF WATER OR 28 INCHES COLUMN.

HOW TALL WAS YOUR BUILDING IN INCHES FROM TOP OF WATER TO UNDERGROUND CRACK AREA ?

FIRE HYDRANT PRESSURE TAKES SPECIAL SEALING FOLKS AND IF IT WAS A FLOW OF WATER YOU HAVE TO MATCH THE PRESSURE PLUS 5 PSI PERHAPS TO STOP THE FLOW FROM THE CRACK. Thinking 14.7 PSI atmospheric pressure does NOTHING when figuring WATER COLUMN AND THEN FIGURE SEA WATER COLUMN WHICH MAY BE DIFFERENT ALTOGETHER !

Atmospheric Pressure 14.7 PSI will support a column of water 33.9 feet high. Understanding 1 psi X 1ft/0.433 psi = 2.3 ft (or 28 inches)

Now measure the exterior upper level of the container where water TOP might be then figure in inches then divide by 28 to get pounds per square inch perhaps at the slit in the cement in the pit. Hope that illustrates what your problem may be. NOW remember apply ONLY ENOUGH PRESSURE on the rubber via the plate of steel and screws to STOP THE LEAK. Then pour the hydraulic cement.

NAVAL DAMAGE CONTROL HAVE THE TRAINING, FILMS AND EQUIPMENT TO STOP THAT LEAK IN 2 HOURS FLAT !

JAPANESE NAVY AND U.S. NAVY ALL PRACTICE THIS PROBLEM, ESPECIALLY ON SUBMARINES !

Ralph Charles Whitley, Sr. CFC032631
Backflow Prevention, Inc.
Phone: 813-286-2333
040311 @ 12:32 PM Eastern

-------Original Message-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 4/3/2011 12:05:57 PM
To: cgjak@alaska.com; info@cgjapanatlanta.org; infocul@cgjbos.org; ryoji@japancc.org; jic@japancc.org; cgjd-pr@quest.net; japaninfo@ggjsf.org; info@cgjapansea.org; jet@ws.mofa.go.jp; APFN; APFN-1; tilo@socom.mil
Cc: POTUS_Office Of The President; mailtothechief@cnn.com; Senator Bill Nelson; Rep. Paul; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov; rick.scott@eog.myflorida.com; NRC Allegation
Subject: NEW EMERGENCY REVIEW FOR JAPAN NUCLEAR POWER PLANT CRISIS --- Deep Sea Water Temperatures for Cooling Rods Spent or Used. Special HYDRAULIC CEMENT to seal leak after Salt Water removed from container THEN DAMAGE CONTROL TECHNIQUE FOLLOWED SEALING THE LEAK BEFORE POURING CONCRETE OR HYDRAULIC CEMENT..

LADIES AND GENTLEMEN:

NOT TOO MANY OF YOU NUCLEAR POWER PLANT OPERATORS OR OWNERS HAVE WORKED WITH FLOWING WATER FROM CRACKS IN EARTH/STEEL BULKHEADS OR EVEN POOLS OR CONCRETE ENCLOSURES UNDERGROUND.

Make no mistake, AMERICA and other Nations are at risk should US PROFESSIONALS NOT PROVIDE SOUND VERIFIABLE METHODS TO SOLVE ODD PROBLEMS YOUR NUCLEAR POWER PEOPLE DO NOT ENCOUNTER. Check with any City, State or County SEWER OR WATER DEPARTMENT PROFESSIONALS and they will confirm the information shown below to be sound and will work every single time. Key is only provide enough pressure to STOP THE LEAK not cause further failure of damaged concrete sections but allow extra rubberized seals like one would use on top of canning jars except these will be 1 inch thick but compressible rubber to seal the leak until a full pour can be accomplished with HYDRAULIC CEMENT not average cement.

FLOWING WATER CANNOT BE STOPPED WITH CEMENT POUR EVEN IN THE CONCRETE VAULTS PRESSURE WOULD PUSH IT OUT AND YOU CAN JACKHAMMER THE CEMENT WHICH WILL LIFT AWAY FROM THE FLOW OF CONCRETE THEN PREPARE THE AREA PROPERLY AND APPLY THE PILLOW OF RUBBERIZED MATERIAL THEN PLATE OF STEEL THEN PRESS FROM THE OTHER SIDE AGAINST THE PLATE WITH ITEMS YOU HAVE SEEN NAVAL PERSONNEL FIGHT WATER LEAKING THROUGH HULLS ON SUBMARINES TO SHIPS OF THE FLEET. ASK THE NAVY PERSONNEL WITH DAMAGE CONTROL HOW TO STOP THE LEAK AND THEN POUR HYDRAULIC CEMENT TO STOP ANY FURTHER EXPANSION OF THE CRACK. THEY ARE THE EXPERTS MR. PRESIDENT AND JAPANESE EMBASSY STAFF. USE THEM AND THEIR KNOWLEDGE PLUS THEY MAY EVEN LOAN YOU THE EQUIPMENT SO YOU CAN WATCH A FLICK AND DO IT EVERY TIME !

U.S. NAVY HAS DONE THIS IN DAMAGE CONTROL SCHOOLS AND SHIPBOARD FOR OVER 100 YEARS.

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

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FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER

CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

[[[REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, NO STEEL AS YOU HAVE ANOTHER SIDE WHERE PLATE CAN BE ATTACHED OR HELD IN PLACE WHILE WORKERS ATTACH SPECIAL SCREW BOLTS TO PRESS AGAINST THE PLATE AND RUBBER TO SEAL THE LEAK WITHOUT CAUSING FRACTURE OF MORE. ONCE WATER IS STOPPED THEN AND ONLY THEN FILL THE ENTIRE AREA WITH HYDRAULIC CEMENT IN THAT CONCRETE UNDERGROUND FOUR WALLED CONTAINER USING ONLY FRESH WATER AFTER PUMPING DOWN ALL WATER REMAINING IN THE CEMENT PIT USED FOR ELECTRICAL CONNECTIONS AND FEEDING THE WATER TO THE SEA.

REMEMBER ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS EVEN SCREW JACKS ONLY PUSHING THE PLATE OVER THE PATCH OF RUBBER ENOUGH TO ONLY STOP THE LEAK BEFORE CAUSING MORE PRESSURE CRACKING MORE OF DAMAGED CONCRETE THEN SEAL ALL IN HYDRAULIC CEMENT !

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WATER OR IT WILL FAIL !]]]

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Ralph Charles Whitley, Sr. CFC032631
Backflow Prevention, Inc.
Phone: 813-286-2333
Tampa

040311 @ 12:05 PM Eastern Sunday

Remember:

Storage of spent rods deep in the Salt Water Ocean some 3,000 feet might find the temperature staying at 40 degrees F continuous. Same of your refrigerator at home perhaps. I have been provided with 8 NEW RADStickers from the Inventor Gordhan Patel at J.P. Labs for use by any of your personnel who will be working at the plant and these stickers attach to ID's or Drivers Licenses so the radiation levels can be measured daily, weekly and monthly. Would also recommend placing some eventually on ALL equipment then a log started to see how much radiation the equipment plus personnel are subjected to in the line of work completed at the sites now or in the future. RADTRIAGE units work inside or outside as suggested but the RADSticker can be read after each shift if necessary for workers.

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SIX " IDEA !

As a State of Florida Certified Plumbing Contractor let me tell you that SOLAR ENERGY/DIESEL GENERATORS should be the backup system for batteries and generators running pumps for cooling near ALL NUCLEAR POWER PLANTS AND EARTHQUAKE PROOF BLADDERS AND TANKS SHOULD HOUSE EXTRA FRESH WATER NEAR NUCLEAR REACTORS for just such an emergency. DRY ICE CAN BE DROPPED INTO THE POOLS AS WELL AS LOX OR LIQUID NITROGEN PUSHED THROUGH STEEL PIPING SUBMERGED AND EXPOSED IN THE WATER TO ALLOW COOLING BUT HOSPITALS HAVE LARGE TANKS WHICH CAN BE RELOADED WHEN RUNNING OUT LIKE I FOUND AT TAMPA GENERAL HOSPITAL when working there for 8.5 years.

Why CNN does not show super tankers pulled or pushed loaded with fresh water from even CHINA if necessary into the harbor near the stricken plants or cities to supply fresh potable water for drinking and bathing is still a mystery. 100 foot coils of black plastic piping coiled and attached to tanks with shower handles and ball valve would allow any WORKERS to be decontaminated with fresh warm water or showers but I still like the DECONTAMINATION PROCEDURES WRITTEN BY DR. DOUG ROKKE FOR THE U.S. MILITARY.

Roads should have 100 tankers with fresh water coming into the area like milk trucks to let the people fill gallon zip lock bags which can be doubling as latrine items storing urine and fecal matter to be picked up and burned. COLD WEATHER SIGNALS USE OF DRUM LINERS TWO PER PERSON STEP INSIDE THEN PULL ONE OVER YOUR HEAD AND SLEEP WARM ..... CAMP JAPAN CAN MAKE IT !

Don't start on not having fuel when there are thousands upon thousands of vehicles with gasoline tanks still containing gas sitting everywhere plus trucks and boats using diesel would power super generators which have not arrived for reasons unknown. SNAP LIGHTS or shake and snap which last 12 hours giving off light could be used by people inside rolling latrines over 55 gallon drums with seal top lids LEFT OPEN for pickup on barges and dropping or washing contents off barges at sea to make fish food. People cringed when I recommended taking the dead to barges, having a prayer service then taking barges to sea, chopping up the dead with SUPER STRONG WOOD CHIPPERS then washing down the barges after all are made into FISH FOOD. What are you going to do ? Burn the bodies releasing RADIOACTIVE PARTICLES INTO THE JAPANESE AIR, WATER, SAND AND SOIL OR ON PLANTS AS IT FALLS BACK TO THE GROUND WHILE BODIES ARE CREMATED ? Now that is smart !

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, THEN PLANT OR POUND THICK STEEL PLATES INTO THE GROUND TO ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS PUSHING THE PLATE OVER THE PATCH OF RUBBER ONLY TO STOP THE LEAK BEFORE CRACKING MORE THEN SEAL ALL IN HYDRAULIC CEMENT ! How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !

Think of Japan building a SEAWALL when they pound special plates in the ground then sealing same they POUR CEMENT FOOTERS after sealing the sea water away with pumps DE WATERING the area for a few hours at least. HOW HARD CAN THAT BE TO VISUALIZE.

Now think of 10 mil thick 3 FT SQUARE BAGS then find barges of Concrete in 80 pound bags, Take a few tanker trucks with cement trucks and load the cement trucks with the proper mixture and you have a CEMENT PLANT. Due to the radiation the CEMENT might have to come from inland in those trucks but CEMENT PUMPS like HI-RISE BUILDERS use pouring flooring UP will push the slurry mix any height required and it is CRITICAL to verify FRESH WATER and proper mixture of HYDRAULIC CEMENT to make it set up quickly.

Professionally submitted FREE to the NRC and people of Japan.

Ralph Charles Whitley, Sr. CFC0326321
Tampa Phone 1/813-286-2333 SKYPE: ralphwhitleysr
040211 @ 11:21 AM Eastern Saturday

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Hello Mr. Whitley:

Thank you for your offer to help with the crisis in Japan. It's reassuring to see how helpful and dedicated private citizens have been in light of this disaster.

At this time, the NRC is not accepting volunteers, but you may want to check with your local Red Cross.

Amy

From: Tobin, Jennifer
Sent: Monday, March 21, 2011 4:14 PM
To: Bonaccorso, Amy
Cc: Deavers, Ron
Subject: RE: Reply to Dr. Doug Rokke comments and information for Dr. Leuren Moret Re: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING ---EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MI

Amy/Ron,

If you read down, he wants to offer himself instead of the lives of young Japanese workers. You could send him the standard response for volunteer helpers. I hope that helps!

Jenny (Tobin) Wollenweber
Export Licensing Officer
Office of International Programs
Office: 301-415-2328

From: Bonaccorso, Amy
Sent: Monday, March 21, 2011 4:10 PM
To: Tobin, Jennifer
Cc: Deavers, Ron
Subject: FW: Reply to Dr. Doug Rokke comments and information for Dr. Leuren Moret Re: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING ---EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MI
Importance: High

Not sure if this person needs a response or not....it seems like a rant.

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr. [mailto:backflow.prevention@verizon.net]
Sent: Saturday, March 19, 2011 12:17 PM
To: dlind49@aol.com; leurenmoret@yahoo.com; cgjak@alaska.com; info@cgjapanatlanta.org; infocul@cgjbos.org; ryoji@japancc.org; info@cgimia.org; jet@embjapan.org; NRC Allegation; POTUS_Office Of The President; mailtothechief@cnn.com; Senator Bill Nelson; Senator John McCain; Rep. Paul
Subject: Reply to Dr. Doug Rokke comments and information for Dr. Leuren Moret Re: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING ---EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MIG...
Importance: High

Dr. Rokke and Dr. Moret:

I have the 28 SEP 90 revised TB 9-1300-278 which was revised in July 1996 perhaps and now we leave it up to the experts such as yourself and Dr. Moret to give advice on decontaminating people and equipment at the plants or even extending times on site with added safety using shields or SPACE SUITS since Japan has similar programs.

Should Depleted Uranium Munitions be used by ALLIES to attack Libya forces then more problems will surface with decontamination from particles.

Explosions involving Nuclear Power plants has a new danger for people and that is also RELEASE OF ASBESTOS OR OTHER FIBER CONTAMINATES WHICH WILL ENTER LUNGS AND LUNGS ENCAPSULATE PARTICLES THEN SCAR TISSUE CUTS DOWN LUNG CAPACITY IN A MESOTHELEOMA TYPE DISEASE LIKE ASBESTOSIS WHICH WILL KILL CHILDREN AND ADULTS SOME 20-30 YEARS LATER.

Having ILD or Interstitial Lung Disease personally from Asbestos Exposure work in the U.S. Navy at Shipyards and in Boiler areas plus Welding in same, I can tell you the loss of lung capacity is like throwing marbles over a fiber rug then trying to blow air through the rug from underneath then increasing the marbles thickness until you can only get a small amount of AIR through the rug to the surface. SMOTHERING is best describing results .... Same applies to particles inhaled and encapsulated by Lung protective measures. Where do you get NEW LUNGS ?

The SPACE SUIT with external tank modifications for use in vehicles IMHO would have allowed placement of many monitors plus even directing flow of water cannons, targeting dosimeters readings with a zoom lens or directing work anticipated since the rods and storage areas are not full of steel shot or lead shot let alone covered with lead sheeting or blankets like hospitals use to stop radiation escaping.

I remember the AIRCRAFT which were amphibious and landed on salt water pulling in a load then flying OFF the water to drop super amounts of water on FOREST FIRES. Too bay those special aircraft were not BROUGHT OVER ON CARRIERS OR BARGES THEN UNLOADED NEAR AN AIRFIELD TO BE USED DROPPING UNLIMITED TONS OF SEA WATER ONTO THE ENTIRE PLANT AREAS to not only load compartments but also to wash off or decontaminate ground and buildings. But WHAT does any 70 year old former Navy and Army Veteran know of such problems and solutions ?

Had the CATTLE AND ANIMALS needed FEED dropped on fields by air drops low level over JAPAN they would have been done. NOW IS THE TIME TO USE AIR DROPS OF FOOD AND WATER, EVEN RAMEN NOODLES AND WATER IN PALLETS ALONG WITH TOILET PAPER, ZIP LOCK BAGS, DRUM LINERS AND OTHER CLEAR PLASTIC BAGS FOR USE GETTING FRESH WATER FROM PLANT LEAVES VIA TRANSPIRATION OR SOLAR STILL TECHNIQUES SINCE THOUSANDS OF CONTAINERS <CLEAR PLASTIC BOXES AND GLASS BOXES> ARE THERE TO USE SOLAR STILL TECHNIQUES WHICH THE KIDS WILL SET UP IN THE SUNLIGHT ALL OVER ON ROOFS, PATIO AND OPEN FIELDS GETTING ALL OF THE FRESH WATER

THEY CAN HANDLE PLUS THE FUEL IN THOSE STRANDED VEHICLES CAN BE PLACED IN CONTAINERS TO BURN WATER CONTAINERS TO MAKE SOUPS.....OR RAMEN NOODLES CAN BE EATEN RAW OUT OF THE PACKAGE ONE PACK PER MEAL WILL KEEP PEOPLE ALIVE !

CAMP JAPAN WILL LEARN TO GET INSIDE PLASTIC CLEAR DRUM LINERS THEN PULL ANOTHER OVER THEIR HEADS AND TUCK COVER OVER BODY COVERINGS TO KEEP WARM IN THIS EMERGENCY !

THE MOST IMPORTANT ITEM STILL IS UNANSWERED RELATIVE TO USE OF IRIDIUM 9555 SAT PHONES AND LAPTOPS TO COUPLE THEM WITH S.P.O.T. GPS LOCATORS TO ALLOW REPORTING OF ALL THE NAMES AND FORMER ADDRESSES OF ALL SURVIVORS SO THEIR FAMILIES CAN RELAX KNOWING THEY ARE SAFE. JAPAN SHOULD HAVE A WEB SITE WITH THIS INFORMATION FOR ALL WHO DESIRE TO SEE THE NAMES OF SURVIVORS TO STOP PEOPLE FROM ANXIETY ATTACKS !

300,000 BODY BAGS OR 6 MIL THICK CONTRACTOR BAGS WITH STICK ON ITEMS FOR WRITING NAMES OR IDENTIFICATION WILL BE NECESSARY FOR THIS MASSIVE DISASTER WHICH CAN GO SUPER HIGH IF FRESH WATER IS NOT OBTAINED VIA TANKERS TO THE PLANT OFFSHORE LOCATION THE BARGES CAN BRING THE MATERIAL OVER TO THE SHORE AND DRAW OFF CONCRETE MIXING POINTS FOR PLACING OR PUMPING SEVERAL STORIES HIGH INTO SUPER ROUND TALL STEEL CIRCULAR ITEMS WHICH SHOULD BE IN THE PROCESS AT A SHIPYARD WELDING SHOP AND TRANSFERRED TO BARGES THEN BROUGHT TO SHORE AND MOVED BY HELICOPTER WITH WELDERS OR PERHAPS SPECIAL HINGES MADE IN THE SHIPYARD ALLOWING THEM TO BE SET AROUND BUILDINGS SEALING THEM WITH CONCRETE AS FAR OUT AND AS HIGH AS THEY DESIRE.

OLD PLUMBING TRICK ! EVEN wood with steel straps can do the same thing and be light enough a large helicopter can set it down on the earth then concrete pouring started as STEEL RODS placed 3 or 4 feet in length into ground will allow pushing down and keeping it in position. HINT !

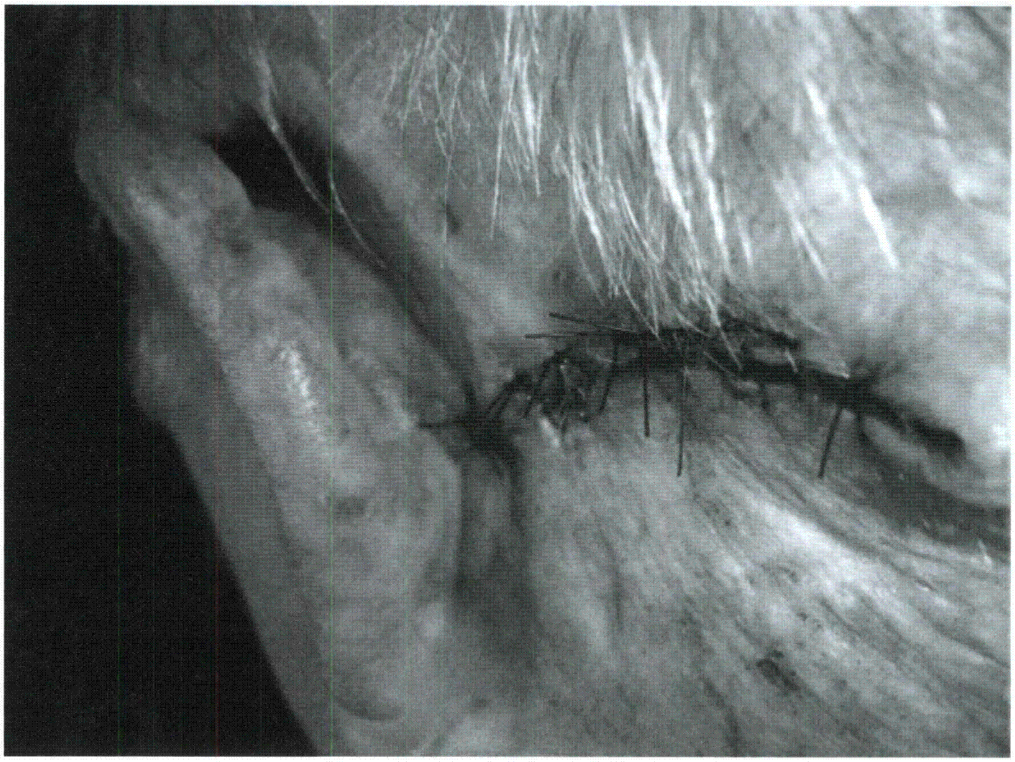
TWO MEN or TWO WOMEN can take a bag of concrete mix, use a specific amount of water, pour into a special 6 mil thick bag and roll the bag back and forth on the ground sealed to allow nothing to escape and it can be used to make a POUR quality amount. Use of SUPER CEMENT TRUCKS with barges of cement or bringing cement from plants to the plant of the same consistency with FRESH WATER MIX would allow dropping loads into pumper units run by generators like used in giant building footers and decks pouring even at several stories high.....with the form in place the cement could be poured making a round or square cement containment which once set up should smother any FIRE and the only problem would be since no rebar is used the water must be fresh and with sea water on rods that must be replaced with FRESH WATER or the items could really stop the pour or cause explosions if gasses built up.

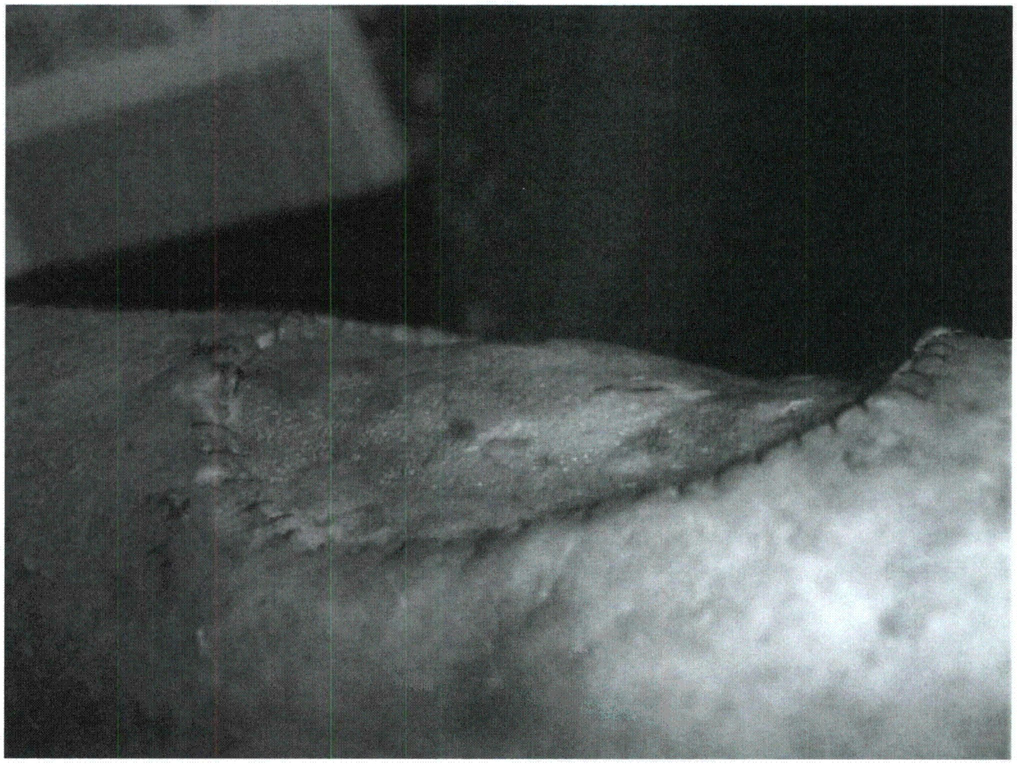
I prefer to think steel shot or lead shot pellets should be dropped IF lead and steel would help then pour fresh water and see if the situation can be corrected and IF NOT sealing with CONCRETE MIGHT BE THE ONLY OPTION but remember the warning on SALT WATER MIXING WITH CONCRETE WILL NOT MAKE IT WORK SO LEAD OR STEEL SHOT 4-6 FEET THICK MIGHT BE REQUIRED BEFORE ADDING PLASTIC PELLETS TO SEAL STEEL FROM FRESH CONCRETE SUPER HEAVY 5,000 PSI HYDRAULIC CEMENT QUALITY WHICH WILL SET UP IN SOME WATER.

Instant communications via CB RADIO with SPEAKER and Magnetic Mount Antenna is my best form of communications for many miles using one on top of a building. Fresh water from WELLS went way over the heads of people in Japan with the agricultural wells not even being considered and quite frankly ONLY YOU have sent back anything direct to me from all of the messages forwarded Dr. Rokke. Tells me something about the OTHER PROFESSIONALS IN JAPAN AND OUR GOVERNMENT.

Sadly people may think of using Salt Water for mixing concrete and pouring or pumping same into any containers which could be round and lightweight to hold the concrete but again SALT WATER DOES NOT MIX WELL WITH CONCRETE AND MAY STOP IT FROM SETTING UP PROPERLY CAUSING FRACTURES WHICH COULD BE DEADLY. That said we have backed off sending any advice to the embassy staff at phone 202-328-2187 FAX or 202-328-2184 FAX lines for Japanese Embassy in Washington nor do we communicate with the Embassies via Internet cgjak@alaska.com info@cgjapanatlant.org infocul@cgjbos.org ryoji@japancc.org infor@cgima.org jet@embjapan.org since NO ONE ANSWERS !

As one recovering from Cancer Surgery with an huge 4 inch X 4 inch deep wound with skin graft on left forearm PLUS neck cutting area on left ear behind ear I am limited and in pain or WE would have tried to come to the site. The Left Thigh 4 x 4 area where they took the right arm graft is still healing after surgery 03/28/11 and constant daily changes of bandages might prevent this 70 year old from being on any crews .... Have SKYPE can be there visually but no one called or asked.





My personal pain level and medications from the James A Haley VA HOSPITAL would
NOT stop me from volunteering to assist as at 70 years old I could replace someone YOUNGER working at or near the plant who might receive a fatal dose of radiation.

Ralph Charles Whitley, Sr.
Tampa 813-286-2333
SKYPE: ralphwhitleysr

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031911 @ 12:15 PM Eastern

-------*Original Message*-------

From: dlind49@aol.com
Date: 3/18/2011 1:56:14 PM
To: backflow.prevention@verizon.net
Subject: Re: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING --- EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MIGHT HELP IN JAPAN......Re: SEA WATER SPRAYED BY FIRE PROTECTION PRESSU

the personal decon procedures are in tb 9-1300-278 you should have that from before

doug

-----Original Message-----
From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr. <backflow.prevention@verizon.net>
To: rick.scott@eog.myflorida.com; jennifer.carroll@eog.myflorida.com; CFO Jeff Atwater <CFO.Jeff.Atwater@myfloridacfo.com>; pam.iorio@tampagov.net; tampacitycouncil@tampagov.net; Beckner, Kevin <BecknerK@HillsboroughCounty.ORG>; higginbothama@hillsboroughcounty.org
Sent: Fri, Mar 18, 2011 9:36 am
Subject: Fw: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING ---EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MIGHT HELP IN JAPAN......Re: SEA WATER SPRAYED BY FIRE PROTECTION PRESSU

FYI

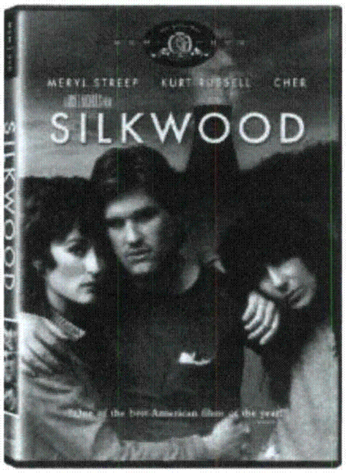
-------*Original Message*-------

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
Date: 3/18/2011 9:23:46 AM
To: mailtothechief@cnn.com; POTUS_Office Of The President; Senator Bill Nelson; Senator John McCain; Rep. Paul; APFN; APFN-1
Cc: tilo@socom.mil; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov;

shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov; Gordhan N. Patel; Albrodsky@aol.com
Subject: FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER....Re: REMOTE MONITORS FOR RADIATION AT PLANT SAT TRANSMITTING --- EMERGENCY REVIEW - SPACE SUIT FOR SPACE WALK MIGHT HELP IN JAPAN......Re: SEA WATER SPRAYED BY FIRE PROTECTION PRESSURE UNITS CAN GO TO THE TOP EASILY IN JAPAN

Ladies and Gentlemen: **FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER !**

RECOMMEND IMMEDIATE VIEWING OF "SILKWOOD" FROM !983 AS YOUR WORKERS ARE NOT PROPERLY DECONTAMINATED DAILY AND SADLY ONLY THE MILITARY MAY KNOW OF THE MEASURES WHICH MUST BE TAKEN......AMERICAN MILIARY KNOW FROM YEARS OF PRACTICE AND WORKING ALL OVER THE PLANET !



The MOVIE AVAILABLE ON NETFLIX NO DOUBT OR DIRECT is the story of Karen Silkwood, a metallurgy worker at a plutonium processing plant who was purposefully contaminated, psychologically tortured and possibly murdered to prevent her from exposing blatant worker safety violations at the plant. Meryl Streep, Kurt Russell and CHER plus others provided a glimpse of what decontamination MUST TAKE PLACE for workers out and about exposed to radiation. Nothing similar is apparently being considered for those brave men and women who work in the plant NOR are the proper DOSIMETER'S WORN OR IN PLACE TO REALLY ALLOW READINGS DAILY AT ALL SITES !

Professionally I have attempted to contact members of the Government of the United States plus Japan with little or no effect as not one PALLET OF WATER, COKE, DR. PEPPER, FOOD, TOILET PAPER, DISPOSABLE CLOTHING, CLEAR PLASTIC DRUM LINERS AND MEDICINES HAVE APPARENTLY BEEN AIR LIFTED AND PARACHUTE DELIVERY DROPPED LOW ALTITUDE ON THOUSANDS OF ACRES OF FARM LAND TO BE PICKED UP AND DISTRIBUTED TO THE PEOPLE.

AMERICAN AND JAPANESE SPACE WALK SUITS WOULD PROTECT WORKERS !
SAME SPACE SUIT MODIFIED TO ACCEPT TANKS OF

OXYGEN ON VEHICLES WOULD ALLOW LONG STAYS OUT IN SUSPECTED CONTAMINATION AREAS THEN RECORDINGS TAKEN BUT DECONTAMINATION MUST OCCUR BY FIRE TRUCK SPRAYING WATER ON EQUIPMENT AND SPACE SUITS BEFORE SAME ARRIVE AT DECONTAMINATION AREAS, GET OUT AND GO THROUGH SPECIAL PROCEDURES TO CHANGE CLOTHING, WASH, BE CHECKED AND THEN GO TO SLEEP BEFORE THE NEXT CREW ENTER, REMOVE STREET CLOTHING AND DON SPACE WALK SUITS LEFT FROM THE PRIOR CREWS AND PROCESS KEEPS GOING OVER AND OVER ALLOWING ALL TO DRIVE AWAY FROM PLANT LONG DISTANCES TO DECONTAMINATION AREAS LIKE MILITARY DO HERE IN AMERICA !

NOT EVEN AMERICAN OR JAPANESE SPACE WALK SUITS HAVE BEEN ISSUED TO WORKERS FOR ASSISTING ON THE OUTSIDE OF THE PLANTS AS I BELIEVE THOSE SUITS CAN BE MODIFIED TO ACCEPT AND USE TANK AFTER TANK OF OXYGEN STRAPPED TO EVEN LARGE GOLF CARTS IF NOT OPEN JEEPS OR HUMVEE TO ALLOW PEOPLE TO GO FOR LONG PERIODS INTO THE REACTOR AREAS TO ATTACH DOSIMETERS, READ RADIATION, PLANT MORE DOSIMETERS TO TRANSMIT READINGS IF NECESSARY THEN THOSE SAME PEOPLE COULD RETURN TO SPECIAL AREAS TO BE WASHED DOWN WITH SOAP AND WATER, TAKEN THROUGH SECURE DECONTAMINATION WHERE THEY ARE SCRUBBED WITH FRESH WELL WATER, SOAPED AND DRIED WITH TOWELS GOING INTO SEALED RADIATION CONTAINERS, CHANGE ALL CLOTHING AND SHOES OR BOOTS WHICH ARE THEN BAGGED IN SPECIAL CONTAINERS, ISSUED NEW STREET CLOTHES LIKE DOCTOR GOWNS, TAKEN TO SLEEP OR EAT AND REST AREAS THEN GO THROUGH THE SAME PROCESS IN REVERSE PREPARING FOR THE NEXT DAYS WORK. SPACE SUITS WITH BACKPACKS AND CONNECTORS ALLOWING USE OF LARGE OXYGEN CONTAINERS STRAPPED TO OPEN JEEPS OR HUMVEES TO ALLOW THEM TO GO BACK INTO THE WORK AREA DAILY DECONTAMINATED !

SADLY THE FRESH WATER TANKERS FROM EVEN 50 MILES AWAY ARE NOT RUMBLING IN TO THE DISASTER AREAS SUPPLYING UNLIMITED DRINKING WATER FOR THOSE RESIDENTS TRAPPED IN AREAS WITH LITTLE OR NO FOOD. DEEP WATER WELLS, PALLET AFTER PALLET OF CANNED FOOD, SODA, TOILET PAPER, PLASTIC OR PAPER CLOTHING THROWAWAY TYPES COULD BE ON ALL LOCATIONS. DRUM LINERS 55 GALLON WITH 60 OR MORE PER CONTAINER COULD ALLOW PEOPLE TO STEP INTO ONE DRUM LINER CLOTHED, PLACE ANOTHER DRUM LINER OVER THEIR HEAD AND BE PROTECTED FROM THE COLD WEATHER AND ABLE TO SLEEP WITHOUT TAKING AWAY OXYGEN. WHERE ARE THESE NEEDED SUPPLIES FOR THOSE ISOLATED FROM TOKYO NEAR THE EVACUATED PEOPLE TAKING REFUGE AND WHERE IN THE HELL ARE THE AIR DROPS FROM CHINA, KOREA, AMERICA AND YES EVEN JAPAN LET ALONE TRUCK AFTER TRUCK OF SUPPLIES AND WATER CRITICAL TO

EVERYONE IN THE AREA.

MY RECOMMENDATIONS FOR PLANT TRANSPIRATION, SOLAR STILL, CB RADIO WITH MAGNETIC MOUNT ANTENNAE AND SPEAKERS TO PLAY MESSAGES HOT WIRED OR USED WITH CIGARETTE LIGHTER POWER CONNECTIONS TO PLAY MESSAGES AND MUSIC ARE NOT THERE FOLKS. PEOPLE NEED MEDICINES, AREAS NEED TO GET LISTS OF MEDICINES AND WITH GPS S.P.O.T. LOCATORS THE HELICOPTERS CAN DROP MEDICINES ORDERED BY PHYSICIANS BEFORE THE PEOPLE REALLY START DYING AND YOU HAVE TO INCREASE THE BODY BAGS TO 1,000,000 OR MORE SIMPLY BECAUSE THE PEOPLE DO NOT HAVE FOOD, WATER, MEDICINES AND WHO IN THE HELL IS NOT COLLECTING THE GASOLINE PUMPING FROM STRANDED VEHICLES INTO TANKERS OR 5 GALLON GAS CANS TO ALLOW GENERATORS TO BE RUNNING THESE TEMPORARY FACILITIES HOUSING MANY THOUSANDS OF PEOPLE.

KEEP IN MIND BODIES CAN WASH WITH SEA WATER AND SPECIAL SOAPS ARE AVAILABLE TO KEEP PEOPLE CLEAN. COMMODES AND URINALS CAN BE REPLACED WITH LATRINES ON WHEELS OVER 55 GALLON CONTAINERS WHICH CAN BE MOVED ON WHEELS. FREE TOILET PAPER, SPECIAL SOLAR OR BATTERY POWER LIGHTING OR EVEN SNAP LIGHTS WOULD ALLOW PRIVACY AREAS WHERE MEN, WOMEN AND CHILDREN COULD USE THE LATRINES, GO TO THE AREAS AND WASH THEIR HANDS AND BODIES PLUS CHANGE INTO PAPER CLOTHING DISPOSABLE UNTIL EVERYTHING GETS BACK TO NORMAL AND THE CLOTHING CAN BE BAGGED WITH DRUM LINERS, MARKED WITH NAMES AND IDENTIFICATION WHILE THE PEOPLE HAVE ACCESS TO IRIDIUM 9555 SAT PHONES WHICH ALLOW COMPUTERS TO HOOK UP AND TRANSMIT VIA LAPTOP THEIR EXACT LOCATION TO THE JAPANESE AUTHORITIES WHO CAN SET UP A WEB SITE FOR ADDING NAMES OF THOSE UNABLE TO COMMUNICATE WITH FAMILY MEMBERS WORLDWIDE. HERE IN AMERICA I WOULD PRAY THIS GOVERNMENT WOULD RELAX ALL REGULATIONS AND ALLOW DISH NETWORK TO SET UP MOBILE CONNECTIONS TO THE INTERNET TO ALLOW TV AND INTERNET USE BY LAPTOP COMPUTERS.

BEFORE YOU RULE OUT CB RADIO 40 CHANNEL WITH MAGNETIC MOUNT ANTENNAE AND SPEAKERS FOR HAILING OR ANNOUNCEMENTS DO REMEMBER ONE UNIT ON THE HIGHEST BUILDING POSSIBLE CAN ZONE AN ENTIRE CITY FOR TRANSMISSION WELL OVER 5 MILES IN EACH DIRECTION. CHECK WITH JAPANESE RADIO AMATEUR FOLKS WHO USE SAME PLUS CB SINCE 1950'S.

AGAIN. NO ONE HAS COMMUNICATED NOR ASKED ANY QUESTIONS AND VERY LITTLE INDICATIONS ANYONE IS RECEIVING THESE MESSAGES SO THIS WILL INDEED BE MY FINAL MESSAGE SENT ON THE PROBLEMS IN JAPAN !

Remember that LEAD PELLETS like in shotgun shells were filled with, which are still available to seal the reactor areas, could have been poured into any area to seal radiation perhaps but I still pray you people will start bringing FRESH WATER from lakes, rivers, streams, UNDERGROUND IRRIGATION WELLS to give the people and plants fresh water.

Good LUCK with protecting your plants, workers and those living IN JAPAN because someone is not reading every message sent then RESPONDING YOU HAVE RECEIVED SAME and checking with NRC or Japan Equivalent to the Nuclear Regulatory Commission and MILITARY on same. No one LOOKED AT SILKWOOD IN THEIR LIFETIME ? NO ONE EVER SAW A WATER WASHDOWN SYSTEM ON A DESTROYER !

Visit the SITE IDENTIFIED BELOW and see what Dr. Leuren Moret and Dr. Doug Rokke say about your RADIATION EXPOSURE PROBLEM !

Ralph Charles Whitley, Sr. CFC032631
Tampa, Florida
031811 @ 9:23 AM Eastern

Go visit www.apfn.org/apfn/du.htm I donated this ITEM to another Veteran !

FINAL MESSAGE FROM A FORMER U.S. NAVY NBC WARFARE PETTY OFFICER !

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.

Ralph Charles Whitley, Sr. CFC032631
Backflow Prevention, Inc.
4532 W. Kennedy Blvd. PMB-276
Tampa, Florida 33609-2042 USA
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SCRIBID ID: ralphwhitleysr
SKYPE ID: ralphwhitleysr

SCRIBD WEB PAGE: http://www.scribd.com/ralphwhitleysr

From: Ruth DeLaMater Bundy or Ralph Charles Whitley, Sr.
To: NRC Allegation; mailtothechief@cnn.com; POTUS Office Of The President; Senator Bill Nelson; tilo@socom.mil; rick.scott@eog.myflorida.com
Cc: Harry Smith; Harry Lee Coe; betty.reed@myfloridahouse.gov; CristV@hillsboroughcounty.org; dana.young@myfloridahouse.gov; darryl.rouson@myfloridahouse.gov; greg.steube@myfloridahouse.gov; james.grant@myfloridahouse.gov; janet.cruz@myfloridahouse.gov; jim.boyd@myfloridahouse.gov; joyner.arthenia.web@flsenate.gov; latvala.jack.web@flsenate.gov; rachel.burgin@myfloridahouse.gov; rachel@willienelson.com; rich.glorioso@myfloridahouse.gov; seth.mckeel@myfloridahouse.gov; shawn.harrison@myfloridahouse.gov; storms.ronda.web@flsenate.gov; will.weatherford@myfloridahouse.gov
Subject: WHICH REACTOR IS LEAKING ? USE A FEW GALLONS OF FOOD DYE IN VARIOUS COLORS INSERTED IN THE REACTOR BY COLOR.
Date: Monday, April 04, 2011 9:49:04 AM
Attachments: SENDER_EMAILbackflow@@prevention@verizon@@net.png
Importance: High

FYI. Problem can go INTERNATIONAL quickly then those with tickets on the next NASA FLIGHT may have to land on the MOON FOREVER.

Why they do not use a few gallons of RED FOOD DYE to see which reactor is leaking that water is beyond me. Plumbers USE FOOD DYE to see if a commode is leaking, should work easily on boiling water in any plant....DUMP AND RUN then wait to see if the concrete pit shows RED, GREEN, BLUE.....ORANGE..... Then they locate the leaking item for further work.

SEE: http://en.wikipedia.org/wiki/Food_coloring BRILLIANT BLUE, BRILLIANT RED, BRILLIANT GREEN, ORANGE....four choices at least.

Still believe that trench wall can be sealed or shored up with 1 inch thick rubber mat steel plate and pressure from the other side. THOUSANDS OF CARS AND TRUCKS means thousands of JACKS WITH HANDLES AVAILABLE INSTANTLY plus cutting bars of steel to put in that space before jacking pressure UNTIL IT STOP LEAKING ONLY. Then wrap everything with plastic and POUR HYDRAULIC CEMENT TO SEAL ! Hard....yes in radiation. Will it work to show leaking section or Nuclear Reactor leaking by COLOR A B S O L U T E L Y ! Works every day in America !

County CREWS and CITY SEWER AND WATER CREWS ARE EXPERTS IN SEALING LEAKS IN CONCRETE !

So are Military DAMAGE CONTROL SPECIALISTS ! Military RADIATION MONITOR SPECIALISTS ARE IN JAPAN. Hope the SUBMARINE and FLEET DAMAGE CONTROL PEOPLE ARE ASKED !

NOT MY JOB ! HAVE SKYPE CAN TRAVEL VIA INTERNET IF CALLED OR NOTIFIED OF TIME EASTERN TO BE ONLINE

My skype account is ralphwhitleysr !

Ralph
040411 @ 9:48 AM Eastern T I C K T O C K T I C K T O C K

REMEMBER:

ONE POUND PER SQUARE INCH EQUALS 2.3 FEET HIGH COLUMN OF WATER OR 28 INCHES COLUMN.

HOW TALL WAS YOUR BUILDING IN INCHES FROM TOP OF WATER TO UNDERGROUND CRACK AREA ?

FIRE HYDRANT PRESSURE TAKES SPECIAL SEALING FOLKS AND IF IT WAS A FLOW OF WATER YOU HAVE TO MATCH THE PRESSURE PLUS 5 PSI PERHAPS TO STOP THE FLOW FROM THE CRACK. Thinking 14.7 PSI atmospheric pressure does NOTHING when figuring WATER COLUMN AND THEN FIGURE SEA WATER COLUMN WHICH MAY BE DIFFERENT ALTOGETHER !

Atmospheric Pressure 14.7 PSI will support a column of water 33.9 feet high. Understanding 1 psi X 1ft/0.433 psi = 2.3 ft (or 28 inches)

Now measure the exterior upper level of the container where water TOP might be then figure in inches then divide by 28 to get pounds per square inch perhaps at the slit in the cement in the pit. Hope that illustrates what your problem may be. NOW remember apply ONLY ENOUGH PRESSURE on the rubber via the plate of steel and screws to STOP THE LEAK. Then pour the hydraulic cement.

NAVAL DAMAGE CONTROL HAVE THE TRAINING, FILMS AND EQUIPMENT TO STOP THAT LEAK IN 2 HOURS FLAT !

JAPANESE NAVY AND U.S. NAVY ALL PRACTICE THIS PROBLEM, ESPECIALLY ON SUBMARINES !

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

FIRST REQUIREMENT STOP THE LEAK WITH MATTING AND THEN POUR CONCRETE, EVEN HYDRAULIC INSIDE IF DESIRED AND THE LEVEL OF WATER CAN BE BROUGHT DOWN IN THE CONTAINER:

No need to try to worry when one can place even thick rubberized matting over a crack then reinforce the outer crack with steel and use `12 inch wide steel sections pounded into the ground then apply pressure over the plate with jacks before pouring all in concrete. STOP THE LEAK - ABSOLUTELY. Prevent further cracks if there is an explosion MAYBE NOT but then one can use the huge circular container welded in place where you would seal same all around building then pour concrete over the top some 20 feet high and buried some 10 feet in the ground or more sealing everything ! Read what was recommended then check it out immediately.

[[[REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, NO STEEL AS YOU HAVE ANOTHER SIDE WHERE PLATE CAN BE ATTACHED OR HELD IN PLACE WHILE WORKERS ATTACH SPECIAL SCREW BOLTS TO PRESS AGAINST THE PLATE AND RUBBER TO SEAL THE LEAK WITHOUT CAUSING FRACTURE OF MORE. ONCE WATER IS STOPPED THEN AND ONLY THEN FILL THE ENTIRE AREA WITH HYDRAULIC CEMENT IN THAT CONCRETE UNDERGROUND FOUR WALLED CONTAINER USING ONLY FRESH WATER AFTER PUMPING DOWN ALL WATER REMAINING IN THE CEMENT PIT USED FOR ELECTRICAL CONNECTIONS AND FEEDING THE

WATER TO THE SEA.

REMEMBER ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS EVEN SCREW JACKS ONLY PUSHING THE PLATE OVER THE PATCH OF RUBBER ENOUGH TO ONLY STOP THE LEAK BEFORE CAUSING MORE PRESSURE CRACKING MORE OF DAMAGED CONCRETE THEN SEAL ALL IN HYDRAULIC CEMENT !

How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !]]]

Professionally submitted FREE to the NRC and people of Japan Sunday April 3, 2011 as I fear not giving you the information could risk MASSIVE NUCLEAR EXPLOSIONS AT THE PLANT AND ENDANGER AMERICA AS WELL AS THE ENTIRE PLANET.

Have passport and medications for four weeks Mr. President and Japanese Embassy Staff should I be needed there. Japan Airlines can bring me to your Nation if necessary as this inability to follow professional recommendations is unbelievable. WHERE IS THE MAN TO HOLD HIS HAND OVER THE WATER FLOW LIKE PUTTING A FINGER IN A RESERVOIR LEAK ? IS THAT PROFESSIONALISM DROPPING CONCRETE ONTO A FIRE HYDRANT FLOW OF WATER IN A CONCRETE BOX IN THE GROUND ? 1 POUND OF WATER PRESSURE PER HOW MANY INCHES COLUMN OF WATER STORED ? I WAS TAUGHT 14.7 INCHES COLUMNAR EQUALS 1 POUND. How tall is the building and how many pounds of pressure will be pushing water through the crack. THIS IS NOT A FISH TANK FOLKS ! CHECK WITH THE NAVAL SUBMARINE SERVICE AND SHIPS OF THE NAVY....JAPANESE NAVY GOES THROUGH SIMILAR TRAINING BUT YOU HAVE A BASE IN JAPAN PLUS FLEET OFFSHORE TO DRAW INFORMATION FROM..... BORROW THEIR EQUIPMENT AND SCREW JACKS FOR THAT CEMENT SECTION.

Remember:

Storage of spent rods deep in the Salt Water Ocean some 3,000 feet might find the temperature staying at 40 degrees F continuous. Same of your refrigerator at home perhaps. I have been provided with 8 NEW RADStickers from the Inventor Gordhan Patel at J.P. Labs for use by any of your personnel who will be working at the plant and these stickers attach to ID's or Drivers Licenses so the radiation levels can be measured daily, weekly and monthly. Would also recommend placing some eventually on ALL equipment then a log started to see how much radiation the equipment plus personnel are subjected to in the line of work completed at the sites now or in the future. RADTRIAGE units work inside or outside as suggested but the RADSticker can be read after each shift if necessary for workers.

Remember STORAGE OF RODS SPENT OR REMOVED FROM REACTORS means WATER WITH DRY ICE COOLING FOR TRIP TO SEA THEN DROPPING RODS going to 12,000 feet depth in THE BATHYPELAGIC ZONE below the 3,000 feet to say 12,000 FEET might find 36 degrees F. Each is suitable perhaps for storage of spent fuel rods in very deep sections of sea water. Such no doubt was the bulk of laughter when it was recommended that spent rods be deep six stored rather than underground in areas like the scrubbed YUCCA MOUNTAIN storage which might now be for members of Congress and their families to live in any Nuclear Emergency in the USA. Wonder if SUPER THICK MILL PLASTIC SHEATING WOULD ALLOW SEALING FROM SALT WATER YET LET COLD GO THROUGH TO KEEP RODS SAFE ? Check with NRC personnel on this NEW STORAGE METHOD " DEEP SIX " IDEA !

As a State of Florida Certified Plumbing Contractor let me tell you that SOLAR ENERGY/DIESEL GENERATORS should be the backup system for batteries and generators running pumps for cooling near ALL NUCLEAR POWER PLANTS AND EARTHQUAKE PROOF BLADDERS AND TANKS SHOULD HOUSE EXTRA FRESH WATER NEAR NUCLEAR REACTORS for just such an emergency. DRY ICE CAN BE DROPPED INTO THE POOLS AS WELL AS LOX OR LIQUID NITROGEN PUSHED THROUGH STEEL PIPING SUBMERGED AND EXPOSED IN THE WATER TO ALLOW COOLING BUT HOSPITALS HAVE LARGE TANKS WHICH CAN BE RELOADED WHEN RUNNING OUT LIKE I FOUND AT TAMPA GENERAL HOSPITAL when working there for 8.5 years.

Why CNN does not show super tankers pulled or pushed loaded with fresh water from even CHINA if necessary into the harbor near the stricken plants or cities to supply fresh potable water for drinking and bathing is still a mystery. 100 foot coils of black plastic piping coiled and attached to tanks with shower handles and ball valve would allow any WORKERS to be decontaminated with fresh warm water or showers but I still like the DECONTAMINATION PROCEDURES WRITTEN BY DR. DOUG ROKKE FOR THE U.S. MILITARY.

Roads should have 100 tankers with fresh water coming into the area like milk trucks to let the people fill gallon zip lock bags which can be doubling as latrine items storing urine and fecal matter to be picked up and burned. COLD WEATHER SIGNALS USE OF DRUM LINERS TWO PER PERSON STEP INSIDE THEN PULL ONE OVER YOUR HEAD AND

SLEEP WARM ..... CAMP JAPAN CAN MAKE IT !

Don't start on not having fuel when there are thousands upon thousands of vehicles with gasoline tanks still containing gas sitting everywhere plus trucks and boats using diesel would power super generators which have not arrived for reasons unknown. SNAP LIGHTS or shake and snap which last 12 hours giving off light could be used by people inside rolling latrines over 55 gallon drums with seal top lids LEFT OPEN for pickup on barges and dropping or washing contents off barges at sea to make fish food. People cringed when I recommended taking the dead to barges, having a prayer service then taking barges to sea, chopping up the dead with SUPER STRONG WOOD CHIPPERS then washing down the barges after all are made into FISH FOOD. What are you going to do ? Burn the bodies releasing RADIOACTIVE PARTICLES INTO THE JAPANESE AIR, WATER, SAND AND SOIL OR ON PLANTS AS IT FALLS BACK TO THE GROUND WHILE BODIES ARE CREMATED ? Now that is smart !

See http://www.scribd.com.ralphwhitleysr as these items needed NOW in Japan were provided on the Internet for the earthquake in Haiti from a Florida Contractor!

NOW FOR NUCLEAR REACTOR LEAK WITH WATER CONTAMINATION: Ever hear of QUICKRETE HYDRAULIC CEMENT which almost sets up in water which can be poured outside a container area with a crack while de-watering through pumps and special piping with filters occurs keeping the area dry until a few cement trucks can pull up and the entire load forced into a hole which will also cause cement to seal the crack and come inside the container a small amount. Plumbing Contractors USE hydraulic cement for leaks with septic tanks which are concrete and steel. CHECK WITH THE EXPERTS IN JAPAN since WE have to mix only what we will use in 3 minutes so several people with those 10 mil thick bags or super powerful mixing items can allow mixing and pouring quickly into the hole sealing the crack then cement trucks with different cement mixture can apply a concrete WEDGE WITH STEEL SUPPORT PILINGS TO KEEP IT ALL IN PLACE ! Time involved including digging perhaps less than 2 hours on site. HYDRAULIC CEMENT MUST BE PLACED QUICKLY INTO THE AREA that is why we recommended using a rubber section and steel supported by pilings and plate with hydraulic jacks. Gives you a little more time to work out the problems stopping the water flow and allowing de-watering of the soil.

http://www.quikrete.com/ProductLines/HydraulicWaterStopCementPro.asp

http://www.quikrete.com/PDFs/DATA_SHEET-Hydraulic%20Water-Stop%20Cement%201126.pdf

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REMEMBER 1 FOOT THICK RUBBER MAT CUT X FEET SQUARE OVER CRACK IN WALL OR CONTAINER ALLOWING 12 INCHES UP, SIDES AND DOWN FOR FUTURE CRACKS, THEN PLANT OR POUND THICK STEEL PLATES INTO THE GROUND TO ALLOW A FORMED CUT SECTION OF STEEL TO BE LOWERED THEN HYDRAULIC JACKS PUSHING THE PLATE OVER THE PATCH OF RUBBER ONLY TO STOP THE LEAK BEFORE CRACKING MORE THEN SEAL ALL IN HYDRAULIC CEMENT ! How long does that take and you have stopped the leak or crack ! HYDRAULIC CEMENT MUST ONLY BE MIXED WITH FRESH WATER FOLKS......NO SALT WATER OR IT WILL FAIL !

Think of Japan building a SEAWALL when they pound special plates in the ground then sealing same they POUR CEMENT FOOTERS after sealing the sea water away with pumps DE WATERING the area for a few hours at least. HOW HARD CAN THAT BE TO VISUALIZE.

Now think of 10 mil thick 3 FT SQUARE BAGS then find barges of Concrete in 80 pound bags, Take a few tanker trucks with cement trucks and load the cement trucks with the proper mixture and you have a CEMENT PLANT. Due to the radiation the CEMENT might have to come from inland in those trucks but CEMENT PUMPS like HI-RISE BUILDERS use pouring flooring UP will push the slurry mix any height required and it is CRITICAL to verify FRESH WATER and proper mixture of HYDRAULIC CEMENT to make it set up quickly.

Professionally submitted FREE to the NRC and people of Japan.

Lee, Richard

From: Larzelere, Alex [alex.larzelere@nuclear.energy.gov]
Sent: Monday, April 04, 2011 4:29 PM
To: DL-NITsolutions
Subject: Slides for Today's Call
Attachments: image001.jpg; 0404 S-1 Briefing rev 1.pptx

Everyone,

Here is the material for today's call.

Alex

Alex R. Larzelere
Director, Advanced Modeling and Simulation Office
Office of Nuclear Energy (NE-71)
U.S. Department of Energy
202-586-1906
Alex.Larzelere@nuclear.energy.gov

AMSO
ADVANCED MODELING & SIMULATION OFFICE

4/3/14

Coyne, Kevin

From: Coyne, Kevin
Sent: Tuesday, April 05, 2011 12:31 PM
To: Stutzke, Martin
Cc: Hudson, Daniel; Correia, Richard
Subject: FW: SOARCA likely to be referenced, questioned tomorrow
Attachments: Level 3 PRA RIC_hudsond-h.pdf

Importance: High

Marty -

I have attached the Level 3 RIC presentation. If you could add a bit more commentary and context to the following bullets and forward to Rich I'd very much appreciate it (e.g., is there any context to add to WASH-1400, NUREG-1150, etc...). Also feel free to revise, edit, or collapse the bullets:

- A Level 3 Probabilistic Risk Assessment (PRA) considers:
  - A range of initiating event categories (e.g., fires, flooding, seismic, and plant equipment failures)
  - Plant response to postulated
  - Core damage progression
  - Radiological release, weather, evacuation, and public health consequences
  - Goal is to quantify risk in a systematic manner

- Prior studies estimating nuclear power plant risk to public
  - WASH-740 (March 1957)
  - WASH-1400 (October 1975)
  - NUREG-1150 (December 1990)

- NRC staff initiative for a comprehensive site Level 3 PRA based on:
  - PRA and technical advances since NUREG-1150
  - Interest in site accident risk versus reactor accident risk
  - Desire to use a more integrated and consistent analysis approach
  - Enhance NRC staff PRA capability by developing in-house risk expertise

- Commission tasking (SRM M100218)
  - Engage internal and external stakeholders in formulating plan and scope for future actions
  - Commission provided conditional support for Level 3 PRA related activities
  - Requested the staff to provide options for proceeding with Level 3 PRA (staff plans to provide an options SECY paper to Commission in July)

- Potential uses of a Level 3 PRA
  - Inform policymaking and rulemaking
  - Focus NRC's inspection program
  - Resolution of generic safety issues
  - Prioritization of safety research programs

From: Santiago, Patricia
Sent: Tuesday, April 05, 2011 11:31 AM
To: Coyne, Kevin; Stutzke, Martin
Cc: Correia, Richard; Wagner, Katie; Lee, Richard
Subject: FW: SOARCA likely to be referenced, questioned tomorrow
Importance: High

4/315

FYI
I know Dan and Doug are out and wanted to make sure you had the request. It is related to the congressional briefings that Brian has been doing related to Japan.
thanks

From: Sheron, Brian
Sent: Tuesday, April 05, 2011 11:24 AM
To: Santiago, Patricia; Correia, Richard
Cc: Uhle, Jennifer; Gibson, Kathy
Subject: FW: SOARCA likely to be referenced, questioned tomorrow

See below. Can I get some background bullets on SOARCA and level 3 PRA within a couple of hours?

From: Rihm, Roger
Sent: Tuesday, April 05, 2011 11:17 AM
To: Sheron, Brian
Subject: FW: SOARCA likely to be referenced, questioned tomorrow

It seems this hearing is going everywhere. I know you are sending over some material on dry cask storage. Can you also provide a limited amount of background material on SOARCA and level 3 PRAs? I have the one pagers from NUREG 1925 to start with. Thx.

From: Powell, Amy
Sent: Tuesday, April 05, 2011 11:10 AM
To: Virgilio, Martin
Cc: Rihm, Roger; Shane, Raeann; Schmidt, Rebecca; Sheron, Brian
Subject: SOARCA likely to be referenced, questioned tomorrow

Marty –

OCA got a heads up from Mr. Waxman's staff that he and Rep. DeGette may reference the concept of SOARCA, work to date, and ask related questions at tomorrow's hearing. Dr. Sheron did a briefing for a number of House Energy and Commerce staffers that referenced ongoing work on this; staff was impressed so encouraged their bosses to ask about it (understanding that it is evolving, draft, preliminary, etc.).

Amy

Amy Powell
Associate Director
U. S. Nuclear Regulatory Commission
Office of Congressional Affairs
Phone: 301-415-1673

U.S.NRC

RIC 2011
Comprehensive Site Level 3 Probabilistic Risk Assessment (PRA)

Dan Hudson
Office of Nuclear Regulatory Research
March 8, 2011

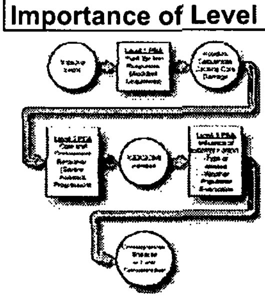
USNRC

Presentation Objectives

- Provide updated information to external stakeholders about this evolving NRC staff initiative.
- Encourage external stakeholder engagement and participation in upcoming activities.

USNRC

Importance of Level 3 PRAs



Risk Characterization:
Level 1 PRA
Level 2 PRA
Level 3 PRA

Key Message:
A Level 3 PRA is required to estimate the integrated risk to the public from all hazards.

USNRC

Historical Perspective

- **Prior studies estimating risk to public**
  - WASH-740 (March 1957)
  - WASH-1400 (October 1975)
  - NUREG-1150 (December 1990)

18 years

15 years

- **PRA Policy Statement** (August 1995)
  - Implementation of risk-informed regulation

Key Message: Even before implementation of risk-informed regulation, the NRC set a precedent for periodically updating its understanding of nuclear reactor accident risk.

4

USNRC

Comprehensive Site Level 3 PRA

- **NRC staff initiative based on:**
  - Advances since NUREG-1150
  - Interest in site accident risk versus reactor accident risk
- **Commission tasking**
  - Engage internal and external stakeholders in formulating plan
  - Provide options for proceeding with Level 3 PRA activities

Key Message: The NRC staff believes it is time to conduct a new site Level 3 PRA to update and improve our understanding of nuclear site accident risk.

5

USNRC

Comprehensive Site Level 3 PRA (cont.)

- **Phase 1 – Scoping Study (FY2010-FY2011)**
- **Phase 2 – Pilot Study (start in FY2012)**
- **Phase 3 – Follow-on studies (as needed)**

Key Message: To optimize cost-benefit, the NRC staff is using a three-phased approach to conducting new Level 3 PRA activities.

6

USNRC

Scoping Study Objectives

- **Develop options for the following aspects of a potential site Level 3 PRA pilot study:**
  - Scope of the analysis and PRA technology to be used
  - Perspectives on future uses of results
  - Site selection attributes
  - Resource estimates
- **Identify NRC staff's recommendation for the pilot study**
- **Obtain external stakeholder support**

7

USNRC

Potential Pilot Study Objectives

- **Update and improve our understanding of nuclear site accident risk by:**
  - Incorporating advances since NUREG-1150
  - Using a more integrated and consistent analysis approach
- **Enhance our PRA capability by:**
  - Integrating and bridging gaps between existing analytical tools
  - Developing risk analysis expertise

Key Message: This initiative is primarily an incremental improvement to existing analytical tools – **not** a large-scale developmental effort.

8

USNRC

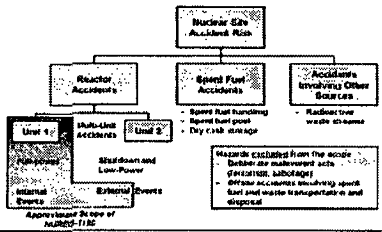
Potential Pilot Study Objectives (cont.)

- **Demonstrate feasibility of conducting lower cost integrated Level 3 PRAs**
- **Evaluate the need for follow-on studies**

Key Message: This initiative is primarily an incremental improvement to existing analytical tools – **not** a large-scale developmental effort.

9

Potential Pilot Study Scope



Key Message: The NRC staff is considering a more complete analysis using a better integrated and consistent approach.

Some Potential Future Uses

- Inform policymaking and rulemaking
- Focus NRC's inspection program
- Resolution of generic safety issues
- Prioritization of safety research programs

Key Message: Much like the NUREG-1150 PRAs, the results of a new site Level 3 PRA may be used to inform a variety of future regulatory activities.

Upcoming Important Activities

- Public meeting (March 21)
- Advisory Committee on Reactor Safeguards (ACRS) Full Committee Meeting (April 7-9)
- Commission paper submission (July 7)

Key Message: External stakeholder engagement and support are needed for this important NRC staff initiative to succeed.

USNRC

Contact Information

Project Manager
Dan Hudson, RES/DRA
Daniel.Hudson@nrc.gov

Work: 301-251-7919
Fax: 301-251-7424

Mail Stop: C4A07M

Technical Monitor
Marty Stutzke, RES/DRA
Martin.Stutzke@nrc.gov

Work: 301-251-7614
Fax: 301-251-7424

Mail Stop: C4A07M

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USNRC

Acronyms and Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
DRA	Division of Risk Analysis
NRC	U.S. Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
RES	Office of Nuclear Regulatory Research
RIC	Regulatory Information Conference

14

Bano, Mahmooda

From:	OECD Nuclear Energy Agency [nea@oecd-nea.org]
Sent:	Tuesday, April 05, 2011 12:32 PM
To:	OECD Nuclear Energy Agency
Subject:	OECD Nuclear Energy Agency: Monthly News Bulletin - April 2011

NEA MONTHLY NEWS BULLETIN

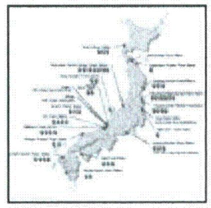
Nuclear Energy Agency

OECD

April 2011 | www.oecd-nea.org

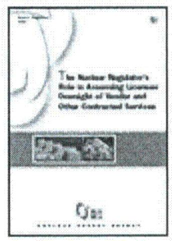
- New at the NEA
- Nuclear safety and regulation
- Radiological protection
- Nuclear law
- Nuclear science
- New publications
- Data Bank

New at the NEA



Responding to the nuclear accident at Fukushima
On 11 March 2011, Japan experienced a major earthquake followed by a tsunami of cataclysmic magnitude. The OECD Nuclear Energy Agency (NEA) wishes to express it condolences to all those who have been affected by this disaster. It has offered its assistance to the Japanese authorities as they address the very challenging situation at the Fukushima nuclear power plant. The NEA will be playing a key role in the evaluation of the accident and the dissemination of lessons learnt based on its various areas of expertise and its competence in addressing emergency and accident management issues. The following updates provide initial insights into some of the steps being taken by the NEA.

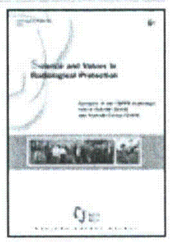
Nuclear safety and regulation



Flashnews activated to share accurate emergency information among nuclear regulators
On 11 March the NEA Working Group on Public Communication of Nuclear Regulatory Organisations (WGPC) activated the Flashnews system in response to the Fukushima accident. Flashnews allows for the fast exchange of information among national nuclear regulators and is used to help inform the public about nuclear events occurring around the world.

New and existing nuclear safety groups consider Fukushima implications
The NEA Committee on Nuclear Regulatory Activities (CNRA) will establish a senior-level task group to exchange information, co-ordinate activities and examine implications in relation to the Fukushima accident. Once established, members of the group will immediately begin exchanging information prior to the first meeting to be held in Paris in early May. The NEA Committee on the Safety of Nuclear Installations (CSNI) will focus on the technical aspects of safety questions raised by the accident. It will identify issues that could require in-depth evaluation by existing or new nuclear safety task groups. The Fukushima accident will be a special topic for discussion during the June CNRA and CSNI meetings and subsequent working group sessions. Please visit the NEA website for more information on nuclear safety.

Radiological protection



INEX-4 and CRPPH meetings present opportunities to discuss Fukushima
The Fukushima accident will have a significant impact on NEA work in radiological protection. A meeting of the Working party on Nuclear Emergency Matters (WPNEM) on May 3-4 that inter alia will discuss the 4th International Nuclear Emergency Exercise (INEX-4) and the annual meeting of the Committee on Radiation Protection and Public Health (CRPPH) on 17-19 May will present the first international opportunities for experts in this field to discuss the preliminary feedback from emergency measures taken in Japan. A further INEX workshop is planned for 6-7 December 2011. During the May meeting, the CRPPH will submit for

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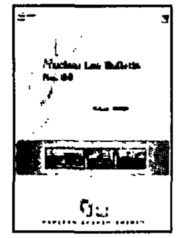
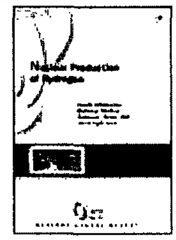
approval a report summarising the resources needed to implement the International Commission on Radiological Protection (ICRP) Publication 60 recommendations into national law and an assessment of the resources that will be needed to implement the new ICRP 103 recommendations. This will provide member countries with information important for implementing these new recommendations as detailed in the International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources. More on NEA work in radiological protection can be found here.

Nuclear law



The legal aspects of the Fukushima accident
NEA Legal Affairs will dedicate a special session of the Nuclear Law Committee (NLC) on 15-16 June to discuss the accident at Fukushima and how the Japanese government intends to deal with liability and compensation for the resulting nuclear damage. In its capacity as secretariat, the NEA is prepared to accommodate discussions on member country initiatives in the field of third party liability for nuclear damage, especially where signatories to the 2004 protocols enhance their efforts for the entry into force of those protocols to provide better protection to potential victims of a nuclear accident. Legal questions related to the accident will be addressed in the June issue of the Nuclear Law Bulletin. Furthermore, the 2011 session of the International School of Nuclear Law will provide an opportunity for the most renowned international nuclear lawyers to exchange on the impacts, lessons learnt and consequences of this accident as it relates to international nuclear law. More information on nuclear law can be found here.

Nuclear science



Nuclear science groups prepared to reassess predictive capabilities
NEA nuclear science working parties and expert groups carry out technical studies in the areas of fuel cycle physics and chemistry, reactor physics, criticality safety, materials performance and radiation shielding. A key focus in each area is on the development, application and validation of modelling tools and their associated nuclear data. These tools are used by the nuclear industry in the design, operation and safety assessment of nuclear facilities including commercial nuclear power plants (NPPs). As details of the Fukushima accident emerge, and as the safety cases and emergency procedures for NPPs are reappraised, NEA nuclear science working parties and expert groups may be required to analyse new scenarios which characterise the evolution of the reactor core and the spent fuel ponds during such an event. Some of these scenarios might challenge the predictive capability of current modelling methods. In that case, new activities could be proposed and discussed by various nuclear science technical groups with the aim of targeting any shortfall in predictive capability, identifying possible methods developments to address the shortfall and providing the means to assess the accuracy of new methods developed. For more information on nuclear science, please visit the NEA website.

New publications

Free publications are available at this link. Paper copies may be requested by sending an e-mail.

The Nuclear Regulator's Role in Assessing Licensee Oversight of Vendor and Other Contracted Services
ISBN: 978-92-64-99157-6, 38 pages.

Publications on sale can be ordered at the OECD bookshop.

Data Bank

NEA Data Bank newsletter

Computer program services

New computer programs available

31-MAR-11	CSNI2017	MCCI-2 PROJECT, Melt Coolability and Concrete Interaction Phase 2 Project (Arrived)
29-MAR-11	NEA-1857	PHITS-2.24, Particle and Heavy Ion Transport code System (Tested)
28-MAR-11	CCC-0295	ELGATL, Calculation of Energy Spectra from Coupled Electron-Photon Slowing Down (Arrived)

22-MAR-11	USCD1240	VIM_NC, VIM color syntax for Nuclear Codes: NJOY, DRAGON, PARTISN, TORT, MONK, and MCNP (Tested)
16-MAR-11	IAEA1287	SHIELD, Monte-Carlo Code for Simulating Interaction of High Energy Hadrons with Complex Macroscopic Targets (Tested)
16-MAR-11	IAEA0970	STOPOW, Stopping Power of Fast Ions in Matter (Tested)
15-MAR-11	USCD1238	ALICE2011, Particle Spectra from HMS precompound Nucleus Decay (Tested)
07-MAR-11	CCC-0767	SWORD 3.2, SoftWare for Optimization of Radiation Detectors (Arrived)
03-MAR-11	NEA-1856	VESTA 2.0.3, Monte Carlo depletion interface code (Arrived)
03-MAR-11	NEA-1210	ZZ HATCHES-19, Database for radiochemical modelling (Tested)

About the NEA
NEA membership consists of 29 OECD countries. The mission of the NEA is to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. It provides authoritative assessments and forges common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development. The information, data and analyses it provides draw on one of the best international networks of technical experts.

Lee, Richard

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Tuesday, April 05, 2011 4:13 PM
To: Lee, Richard
Subject: Re: Call to Japan

No calls scheduled
John E Kelly

From: Lee, Richard (NRC)
To: Kelly, John E (NE)
Sent: Tue Apr 05 15:41:15 2011
Subject: Call to Japan

Hi, John:

Do you a call to Japan today?. If yes, what time is it scheduled for?

Thx, Richard

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Lee, Richard

From: Larzelere, Alex [alex.larzelere@nuclear.energy.gov]
Sent: Tuesday, April 05, 2011 4:30 PM
To: Lee, Richard; Adams, Ian
Cc: Kelly, John E (NE)
Subject: RE: today DOE Science Council and call to Japan

Richard,

Sorry for the delay in my response - my email box is full to overflowing. The call with the Science expert will occur a 5pm EDT today. I am not sure about the call with Japan, but will find out and get an answer out to you as soon as possible.

Regards,

Alex

From: Lee, Richard (NRC)
Sent: Tuesday, April 05, 2011 9:02 AM
To: Adams, Ian; Larzelere, Alex
Cc: Kelly, John E (NE)
Subject: today DOE Science Council and call to Japan

Dear Ian and Alex:

I believe the Science Council call is at 5:00pm for today and tomorrow. I do not when the call to Japan will take place for today and tomorrow. Please let me know.

Thx, Richard

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Lee, Richard

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Tuesday, April 05, 2011 5:39 PM
To: Lee, Richard
Subject: Re: handsout for today meeting

Did you get them
John E Kelly

From: Lee, Richard (NRC)
To: Kelly, John E (NE)
Cc: Binder, Jeff
Sent: Tue Apr 05 17:12:17 2011
Subject: handsout for today meeting

Hi John or Jeff:

Please send me the VGs for today conf. call.

Thx, Richard

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Bano, Mahmooda

From:	RMTPACTSU_INC [RMTPACTSU_INC@ofda.gov]
Sent:	Wednesday, April 06, 2011 9:04 AM
To:	RMT_PACTSU; DART_PACTSU; Fleming, James(DCHA/OFDA) [USAID]; Bartolini, Mark (DCHA/OFDA) [USAID]
Subject:	Japan News (NY Times): U.S. Sees Array of New Threats at Japan's Nuclear Plant

Source: NY Times
April 5, 2011

U.S. Sees Array of New Threats at Japan's Nuclear Plant

By JAMES GLANZ and WILLIAM J. BROAD

United States government engineers sent to help with the crisis in Japan are warning that the troubled nuclear plant there is facing a wide array of fresh threats that could persist indefinitely, and that in some cases are expected to increase as a result of the very measures being taken to keep the plant stable, according to a confidential assessment prepared by the Nuclear Regulatory Commission.

Among the new threats that were cited in the assessment, dated March 26, are the mounting stresses placed on the containment structures as they fill with radioactive cooling water, making them more vulnerable to rupture in one of the aftershocks rattling the site after the earthquake and tsunami of March 11. The document also cites the possibility of explosions inside the containment structures due to the release of hydrogen and oxygen from seawater pumped into the reactors, and offers new details on how semimolten fuel rods and salt buildup are impeding the flow of fresh water meant to cool the nuclear cores.

In recent days, workers have grappled with several side effects of the emergency measures taken to keep nuclear fuel at the plant from overheating, including leaks of radioactive water at the site and radiation burns to workers who step into the water. The assessment, as well as interviews with officials familiar with it, points to a new panoply of complex challenges that water creates for the safety of workers and the recovery and long-term stability of the reactors.

While the assessment does not speculate on the likelihood of new explosions or damage from an aftershock, either could lead to a breach of the containment structures in one or more of the crippled reactors, the last barriers that prevent a much more serious release of radiation from the nuclear core. If the fuel continues to heat and melt because of ineffective cooling, some nuclear experts say, that could also leave a radioactive mass that could stay molten for an extended period.

The document, which was obtained by The New York Times, provides a more detailed technical assessment than Japanese officials have provided of the conundrum facing the Japanese as they struggle to prevent more fuel from melting at the Fukushima Daiichi plant. But it appears to rely largely on data shared with American experts by the Japanese.

Among other problems, the document raises new questions about whether pouring water on nuclear fuel in the absence of functioning cooling systems can be sustained indefinitely. Experts have said the Japanese need to continue to keep the fuel cool for many months until the plant can be stabilized, but there is growing awareness that the risks of pumping water on the fuel present a whole new category of challenges that the nuclear industry is only beginning to comprehend.

The document also suggests that fragments or particles of nuclear fuel from spent fuel pools above the reactors were blown "up to one mile from the units," and that pieces of highly radioactive material fell between two units and had to be "bulldozed over," presumably to

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protect workers at the site. The ejection of nuclear material, which may have occurred during one of the earlier hydrogen explosions, may indicate more extensive damage to the extremely radioactive pools than previously disclosed.

David A. Lochbaum, a nuclear engineer who worked on the kinds of General Electric reactors used in Japan and now directs the nuclear safety project at the Union of Concerned Scientists, said that the welter of problems revealed in the document at three separate reactors made a successful outcome even more uncertain.

"I thought they were, not out of the woods, but at least at the edge of the woods," said Mr. Lochbaum, who was not involved in preparing the document. "This paints a very different picture, and suggests that things are a lot worse. They could still have more damage in a big way if some of these things don't work out for them."

The steps recommended by the nuclear commission include injecting nitrogen, an inert gas, into the containment structures in an attempt to purge them of hydrogen and oxygen, which could combine to produce explosions. On Wednesday, the Tokyo Electric Power Company, which owns the plant, said it was preparing to take such a step and to inject nitrogen into one of the reactor containment vessels.

The document also recommends that engineers continue adding boron to cooling water to help prevent the cores from restarting the nuclear reaction, a process known as criticality.

Even so, the engineers who prepared the document do not believe that a resumption of criticality is an immediate likelihood, Neil Wilmshurst, vice president of the nuclear sector at the Electric Power Research Institute, said when contacted about the document. "I have seen no data to suggest that there is criticality ongoing," said Mr. Wilmshurst, who was involved in the assessment.

The document was prepared for the commission's Reactor Safety Team, which is assisting the Japanese government and the Tokyo Electric Power Company. It says it is based on the "most recent available data" from numerous Japanese and American organizations, including the electric power company, the Japan Atomic Industrial Forum, the United States Department of Energy, General Electric and the Electric Power Research Institute, an independent, nonprofit group.

The document contains detailed assessments of each of the plant's six reactors along with recommendations for action. Nuclear experts familiar with the assessment said that it was regularly updated but that over all, the March 26 version closely reflected current thinking.

The assessment provides graphic new detail on the conditions of the damaged cores in reactors 1, 2 and 3. Because slumping fuel and salt from seawater that had been used as a coolant is probably blocking circulation pathways, the water flow in No. 1 "is severely restricted and likely blocked." Inside the core itself, "there is likely no water level," the assessment says, adding that as a result, "it is difficult to determine how much cooling is getting to the fuel." Similar problems exist in No. 2 and No. 3, although the blockage is probably less severe, the assessment says.

Some of the salt may have been washed away in the past week with the switch from seawater to fresh water cooling, nuclear experts said.

A rise in the water level of the containment structures has often been depicted as a possible way to immerse and cool the fuel. The assessment, however, warns that "when flooding containment, consider the implications of water weight on seismic capability of containment."

Experts in nuclear plant design say that this warning refers to the enormous stress put on the containment structures by the rising water. The more water in the structures, the more easily a large aftershock could rupture one of them.

Margaret Harding, a former reactor designer for General Electric, warned of aftershocks and said, "If I were in the Japanese's shoes, I'd be very reluctant to have tons and tons of water sitting in a containment whose structural integrity hasn't been checked since the earthquake."

The N.R.C. document also expressed concern about the potential for a "hazardous atmosphere" in the concrete-and-steel containment structures because of the release of hydrogen and oxygen from the seawater in a highly radioactive environment.

Hydrogen explosions in the first few days of the disaster heavily damaged several reactor buildings and in one case may have damaged a containment structure. That hydrogen was produced by a mechanism involving the metal cladding of the nuclear fuel. The document urged that Japanese operators restore the ability to purge the structures of these gases and fill them with stable nitrogen gas, a capability lost after the quake and tsunami.

Nuclear experts say that radiation from the core of a reactor can split water molecules in two, releasing hydrogen. Mr. Wilmshurst said that since the March 26 document, engineers had calculated that the amount of hydrogen produced would be small. But Jay A. LaVerne, a physicist at Notre Dame, said that at least near the fuel rods, some hydrogen would in fact be produced, and could react with oxygen. "If so," Mr. LaVerne said in an interview, "you have an explosive mixture being formed near the fuel rods."

Nuclear engineers have warned in recent days that the pools outside the containment buildings that hold spent fuel rods could pose an even greater danger than the melted reactor cores. The pools, which sit atop the reactor buildings and are meant to keep spent fuel submerged in water, have lost their cooling systems.

The N.R.C. report suggests that the fuel pool of the No. 4 reactor suffered a hydrogen explosion early in the Japanese crisis and could have shed much radioactive material into the environment, what it calls "a major source term release."

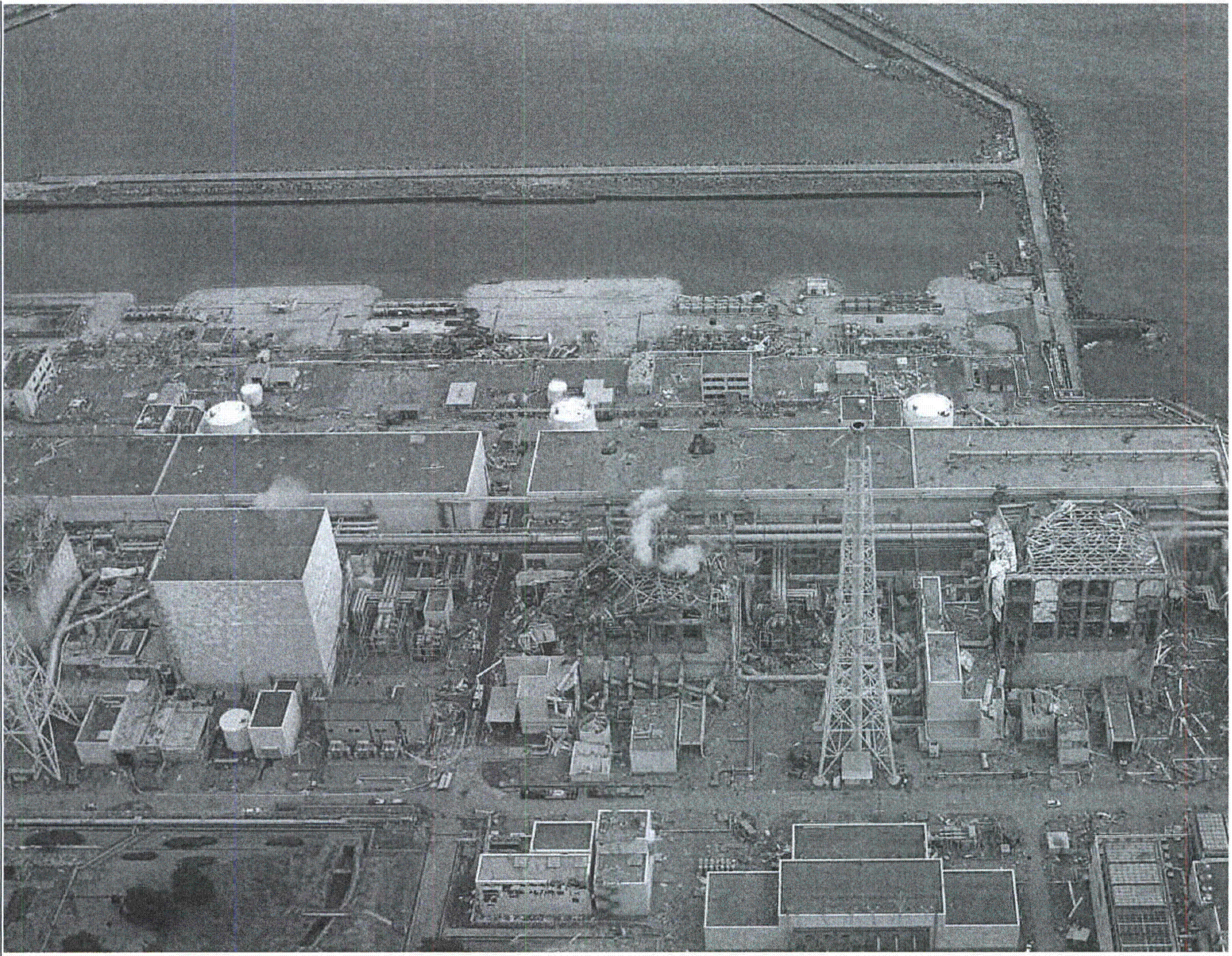
Experts worry about the fuel pools because explosions have torn away their roofs and exposed their radioactive contents. By contrast, reactors have strong containment vessels that stand a better chance of bottling up radiation from a meltdown of the fuel in the reactor core.

"Even the best juggler in the world can get too many balls up in the air," Mr. Lochbaum said of the multiplicity of problems at the plant. "They've got a lot of nasty things to negotiate in the future, and one missed step could make the situation much, much worse."

Henry Fountain contributed reporting from New York, and Matthew L. Wald from Washington.

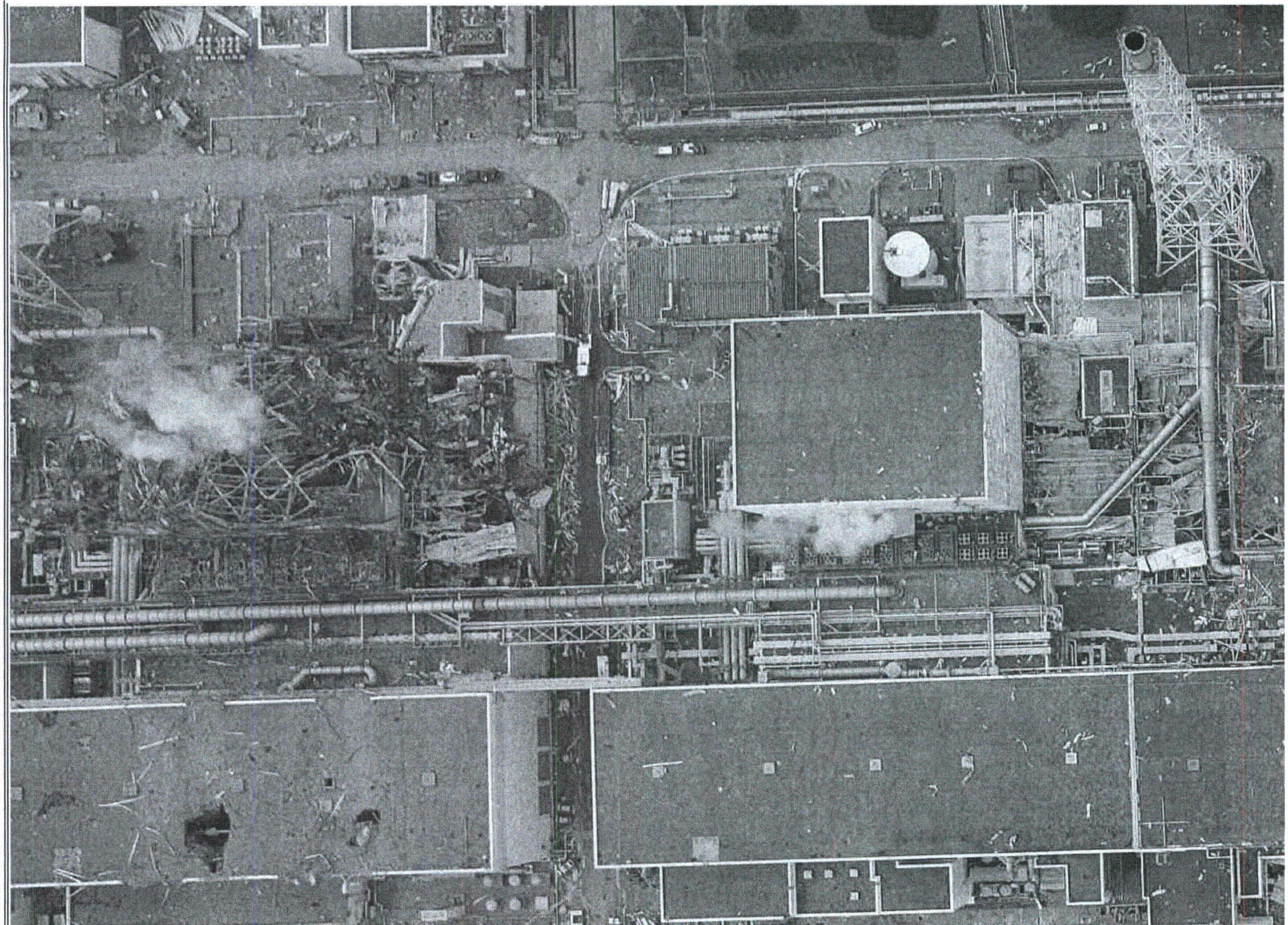
Lily Frey
Information Coordinator

Pacific Tsunami and Japan Earthquake Response Management Team
RMTPACTSU_INC@ofda.gov
202-712-0039



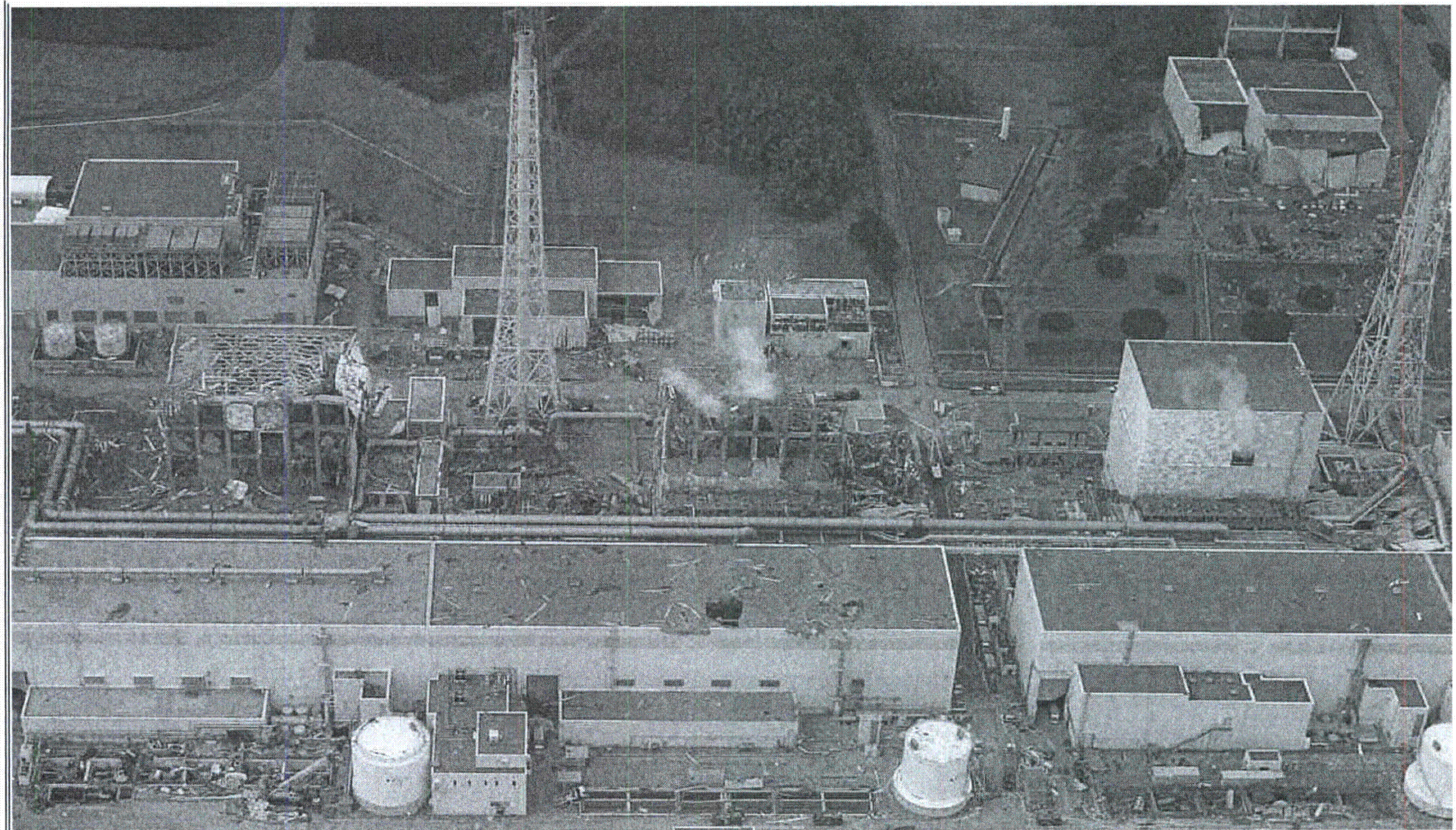
In this March 20, 2011 aerial photo taken by a small unmanned drone and released by AIR PHOTO SERVICE, the crippled Fukushima Dai-ichi nuclear power plant are seen in Okumamachi, Fukush
left: Unit 1, partially seen; Unit 2, Unit 3 and Unit 4. (Air Photo Service Co. Ltd., Japan)

20 March 2011



In this March 20, 2011 aerial photo taken by a small unmanned drone and released by AIR PHOTO SERVICE, the crippled Fukushima Dai-ichi nuclear power plant is seen in Okumamachi, Fukushi right to left: Unit 1, Unit 2 and Unit 3. (Air Photo Service Co. Ltd., Japan)

20 March 2011



In this March 20, 2011 aerial photo taken by a small unmanned drone and released by AIR PHOTO SERVICE, the crippled Fukushima Dai-ichi nuclear power plant is seen in Okumamachi, Fukushi
right to left: Unit 1, Unit2, Unit 3 and Unit 4. (Air Photo Service Co. Ltd., Japan)

From: Helton, Donald
Sent: Wednesday, April 06, 2011 1:29 PM
To: Marksberry, Don
Subject: FW: ti for fukushima

FYI

From: Coyne, Kevin
Sent: Wednesday, April 06, 2011 9:58 AM
To: Helton, Donald
Subject: ti for fukushima

http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf

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Lee, Richard

From: Busby, Jeremy T. [busbyjt@ornl.gov]
Sent: Thursday, April 07, 2011 9:59 AM
To: Binder, Jeffrey L.; Lee, Richard
Cc: 'Doug Burns'
Subject: RE: yesterday DOE conf.call hnadsout
Attachments: 0406 S-1 Briefing rev 1.pptx

Hi Richard,

We'll get you added. Here are the slides from yesterday.

Best regards,
Jeremy

From: Binder, Jeffrey L.
Sent: Thursday, April 07, 2011 9:47 AM
To: 'Richard.Lee@nrc.gov'
Cc: 'Doug Burns'; Busby, Jeremy T.
Subject: RE: yesterday DOE conf.call hnadsout

Doug/Jeremy

Can you get Richard on the list? Thanks.

Jeff

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Thursday, April 07, 2011 09:43 AM Eastern Standard Time
To: Binder, Jeffrey L.
Subject: yesterday DOE conf.call hnadsout

Jeff:

Do you have VGS from yesterday DOE Sci. Council. Conf. call? I do not know what distribution list DOE is using to distribute them ahead of the conf. call. I am not getting it.

Thanks, Richard

4/322

Lee, Richard

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Thursday, April 07, 2011 5:36 PM
To: Lee, Richard
Subject: Re: today's handout

My staff said they sent. Let's test before the meeting
John E Kelly

From: Lee, Richard (NRC)
To: Kelly, John E (NE)
Sent: Thu Apr 07 17:08:14 2011
Subject: today's handout

John:

I have not been receiving any VGs before the meeting for today and the same problems for the previous days. Appreciate it if you can ask someone to send it. This morning, Jeremy sent me the one from yesterday.

Richard

W323

Bensi, Michelle

From: Bensi, Michelle
Sent: Thursday, April 07, 2011 6:25 PM
To: Kauffman, John
Subject: RE: Reminder--OEGIB Weekly Activities Input due by noon tomorrow, Friday 4/8/2011. [eom]

Thanks,
Shelby

Last week activities

- Seismic Q&A document in response to events in Japan
- Presentation (and Prep) for Japan Near-Term Evaluation Task Force Task Force Briefing
- FOIA
- Out-of-office Friday (CHU)

Next week activities

- Seismic Q&A document
- Conference M-W
- Revisions to screening report and other associated misc activities

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From: Kauffman, John
Sent: Thursday, April 07, 2011 7:07 AM
To: Bensi, Michelle; Criscione, Lawrence; Ibarra, Jose; Killian, Lauren; Lane, John; Reisifard, Mehdi; Perkins, Richard; Salomon, Arthur; Smith, April; Wegner, Mary
Subject: Reminder--OEGIB Weekly Activities Input due by noon tomorrow, Friday 4/8/2011. [eom]

W/324

Bensi, Michelle

From: Bensi, Michelle
Sent: Friday, April 08, 2011 12:28 PM
To: Beasley, Benjamin
Subject: public FAQ document

Ben,
With regard to the NRR FAQ document that took seismic questions from an older version of the public FAQ document (which may still be the only one posted online, but is not the most recent version sent to OPA).
I don't have the most updated version of the public FAQs that Annie send to OPA. I will ask her to forward it to you so that you can forward them to NRR to update the seismic questions they pulled from the older version of the document.
Please let me know if you object.
Thanks,
Shelby

W/325

Howell, Art

From: Spitzberg, Blair
Sent: Monday, April 11, 2011 9:11 AM
To: Howell, Art
Subject: RE: HUGE FILE ATTACHED - FUKUSHIMA SLIDE SHOW FROM NISA
Attachments: March 11 tsunami hit reactors.jpg UNable to PRINT

This is the image recently released of the tsunami actually hitting the plant. Note height appears to be above the top of the reactor buildings.

- Blair

From: Howell, Art
Sent: Monday, April 11, 2011 8:04 AM
To: Spitzberg, Blair
Subject: FW: HUGE FILE ATTACHED - FUKUSHIMA SLIDE SHOW FROM NISA

From: Maier, Bill
Sent: Saturday, April 09, 2011 1:12 PM
To: Howell, Art; Howell, Linda
Subject: HUGE FILE ATTACHED - FUKUSHIMA SLIDE SHOW FROM NISA

Art/Linda,

Don't know if you've seen this or not, but it's very comprehensive. 4MB pdf format.

Bill

W/324

Bensi, Michelle

From: Bensi, Michelle
Sent: Monday, April 11, 2011 8:39 PM
To: Kammerer, Annie
Cc: Beasley, Benjamin
Subject: updated public FAQs for NRR

Annie,

As you know, NRR has put together a Sharepoint site with FAQs related to the Japan events. The intent it to make these publically available.

It appears that many of the seismic-related questions came from the public FAQ document that was posted on the public website. I do not think the answers they are using are consistent with the most recent update to the Public FAQs. I believe the older version of the document contains outdated information (e.g. it says that we don't know the GM at the Fukushima plants). I recall that you sent an updated version of the FAQs to OPA, but I don't know if they ever posted it online. Do you think there's value in sending NRR the updated version (if cleared by OPA)?

I don't have the most updated version that you sent to OPA. If appropriate, would you please send the document to Ben (CC'ed on this email) to forward to NRR so that they are using the most updated version of the questions/answers?

Thanks,
Shelby

4/327

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Tuesday, April 12, 2011 4:41 PM
To: Kauffman, John
Subject: FW: List of Issues and Research Areas from Japanese Event
Attachments: Potential Long term Issues Rev1.docx

Please handle this. I will be looking at your draft email on solar storms this afternoon or tomorrow morning.

Ben

From: Correia, Richard
Sent: Tuesday, April 12, 2011 8:55 AM
To: Barnes, Valerie; Beasley, Benjamin; Coe, Doug; Coyne, Kevin; Demoss, Gary; Hudson, Daniel; Ott, William; Peters, Sean; Salley, MarkHenry; Hudson, Daniel; Nicholson, Thomas; Siu, Nathan; Stutzke, Martin
Subject: FW: List of Issues and Research Areas from Japanese Event

All,

Brett Rini has compiled and sorted RES staff input (attached) for the Japan events task force's consideration. Please take a look at his list and annotate any changes/corrections/clarifications keeping in mind how the task force will interpret what will be sending them (i.e., will they understand what we are asking them to consider).

Please send your comments/additions/clarifications back to me and Doug.

thanks

Richard Correia, PE
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
US NRC

richard.correia@nrc.gov

From: Rini, Brett
Sent: Monday, April 11, 2011 5:27 PM
To: Case, Michael; Richards, Stuart; Correia, Richard; Coe, Doug; Gibson, Kathy; Scott, Michael; Valentin, Andrea
Cc: Sheron, Brian; Uhle, Jennifer
Subject: List of Issues and Research Areas from Japanese Event

Division Directors,

Please find attached a list of possible issues and research areas to follow-up on as a result of the Japanese earthquake. I compiled the input I received from your divisions along with a document that Brian sent me and classified the recommendations into various areas (e.g., electrical, severe accidents, external events).

Please review the attached document and let me know if you have any additional thoughts or changes.

Thanks,

Brett

W/328

Brett A. Rini
Technical Assistant
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
(301)251-7615
Brett.Rini@nrc.gov

Potential Long term Issues & Research Areas as a Result of Japanese Earthquake/Tsunami and Impact on Nuclear Power Plants

Electrical Power / Station Black-out

- Assess plant response to long-term loss of onsite and offsite electrical power, as well as capabilities for mitigation (DE)
- Evaluate battery discharge duration when operated under light load (DE)
- Do we need to revisit the need for non-AC dependent hydrogen igniters on IC plants?
- Do we need AC-powered (with battery backup) hydrogen igniters in reactor buildings and/or in the vicinity of SFPs?
- Do plants have EDGs and their associated fuel tanks sufficiently protected from natural phenomena, especially floods?
- Assess the feasibility of licensees developing procedures to bring in portable electric generators to the site to a prepared location, and connecting the generators to the plant electric system. (DE)
- Assess the feasibility of developing procedures to bring in a 125 VDC battery bank and connect it to the plant DC system. (DE)
- Re-assess SBO capabilities at U.S. plants (DE)
- Should SBO coping strategies be seismically qualified to help mitigate beyond design basis seismic events where restoration of offsite power could be delayed beyond the coping time.

Instrumentation & Controls

- Do we have sufficient instrumentation in plants to accurately assess plant conditions following an accident, including severe accidents (e.g., water levels at various locations)? Is the instrumentation sufficiently robust to survive in the accident conditions?
- Is there additional instrumentation that would be of use to help manage a severe accident, such as hydrogen sensors, and would additional measures be necessary to ensure they are viable during a severe accident.
- Consider the need for additional severe accident monitoring instrumentation. Consideration should be given to providing for remote readings from the instrumentation at locations away from the unit. Wireless technology could potentially minimize the cost involved. (DE)
- Reassessment of instrumentation that can provide details on the progression of a severe accident; include remote monitoring of temperatures, pressures and radiation levels using high-capacity (long term) batteries (DE)

Reactor Pressure Vessel & Reactor Coolant System

- Performance issues of degraded/aged components: (DE)
    - Thermal loading: thermal shock, thermal transients
    - Pressure loading: explosive loadings, from thermal transients
- Components/Structures/Materials Performance in Severe or Beyond Design Basis Accidents: (DE)
    - Pumps/Valves
    - Seismic loading
- Weld Residual Stress Compendia: (DE)
    - Database of residual stresses of nuclear components: measurements & model results

- Materials research on the impact of lake/river/sea water used as makeup water to the reactor coolant system and SFP during an accident and impact on subsequent establishment of recirculation. (DE)

Containment

- PWR Containments do not have filtered vents. It is also not clear if they have vents that can be operated without AC power. Consider evaluating the benefits of putting a filtered vent on a PWR containment, along with vents that can be actuated without AC power (e.g. compressed air).
- reevaluate the need for filtered containment venting (DE)
- GSI-191 impact from seawater (DE)
- Assess coatings in the severe accident environment (DE)

Severe Accidents & Mitigation

- Effectiveness of SAMGs and EDMGs provisions (including operator training) (DRA)
- Develop SAMGs that include procedures for a containment breach (DE)
- Assess effects of high general radiation levels from a core melt on the ability for personnel to man control rooms and implement SAMGs (DE)
- Assess the need for additional regulatory guidance for severe accidents (DE)
- Review Severe Accident Management Guidelines/Emergency Operations Plans (DE)
    - Check core and spent fuel cooling procedures
        - Identify any materials issues with the cooling procedures (use of salt/river water in an emergency
- Do U.S. plants have the capability to inject ultimate heat sink water? How much time do plants with cooling ponds, like Palo Verde, have if they injected their ponds? Does that affect long term cooling strategies?
- Emergency H2 venting and whether current US plant configurations could lead to pockets of H2 in areas not covered by H2 igniters or recombiners, that give rise to explosive power sufficient to damage BWR secondary containments. (DRA) Adequacy and placement of hydrogen recombiners/igniters (DE)
- Are there accident management strategies in place for lower vessel flooding, and how well do we understand whether lower vessel flooding will work to retain a molten core inside the vessel?
- Fukushima 3 had several MOX fuel assemblies in it. How would a core with more or a full load of MOX assemblies affect the outcome of severe accidents?

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- Control room staffing and plant staffing for severe accidents (DRA)
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Incident Response / Coordination

- Emergency response given large area wide catastrophe and what can be expected (DRA)
- Assess onsite and offsite responder capabilities at U.S. plants (beyond B.5.b) (DE)
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- It took a while before we called in industry and got an industry consortium going to interact directly with their Japanese counterparts (TEPCO). Should we encourage industry to create a standing consortium that would be poised to move in the event of another accident? Is this really a role for WANO?
- Given overwhelming media interest, define the role of NRC in communicating general information on nuclear energy to the public even if incidents occur at foreign nuclear plants (DE)

Esmaili, Hossein

From: Esmaili, Hossein
Sent: Tuesday, April 12, 2011 5:21 PM
To: Marksberry, Don
Subject: FW: FYI: Staff Presentation hosted by Mike Scott

Maybe we want to attend this tomorrow.

From: Kardaras, Tom
Sent: Thursday, April 07, 2011 12:02 PM
To: RES Distribution
Subject: FYI: Staff Presentation hosted by Mike Scott

Michael Scott, Deputy Director of DSA, will be giving a presentation to RES staff on his travels and experiences while assisting in the Japanese Tsunami/Nuclear disaster from 10 –11am on Wednesday, April 13 in Room 6B-01. In case of overflow in the main conference room, the presentation will also be simultaneously broadcast via VTC in Room 2C-19.

Regards,
Tom Kardaras, Deputy Director (Acting)
Program Management, Policy Development and Analysis Staff
Office of Nuclear Regulatory Research
(o) 301-251-7667

W/329

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Wednesday, April 13, 2011 6:52 AM
To: Kauffman, John
Subject: FW: Some Additional Items
Attachments: Potential Long term Issues Rev1.docx; Potential Long term Issues.docx

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From: Coe, Doug
Sent: Tuesday, April 12, 2011 5:34 PM
To: Barnes, Valerie; Beasley, Benjamin; Coyne, Kevin; Demoss, Gary; Nicholson, Thomas; Ott, William; Peters, Sean; Salley, MarkHenry; Siu, Nathan; Stutzke, Martin
Cc: Correia, Richard
Subject: FW: Some Additional Items

All,
FYI – Here's the consolidated list of possible research topics Brian sent forward to the near-term Task Force this afternoon.

Ben,
See the rev 1 document first category (SBO). Seems like this input to the Task Force pretty well covers the ground you had discussed. If there is something your staff would like to add to this list, please let me know.

Nathan – I've forwarded your input along to Brett Rini with a request to add it.

Mark – the items you forwarded look more like DSA research items. I'll forward them to DSA for consideration.

Doug

From: Sheron, Brian
Sent: Tuesday, April 12, 2011 2:42 PM
To: Miller, Charles
Cc: Holahan, Gary; Grobe, Jack; Dorman, Dan; Sanfilippo, Nathan; Rini, Brett; Weber, Michael; Virgilio, Martin; Case, Michael; Coe, Doug; Correia, Richard; Gibson, Kathy; Richards, Stuart; Scott, Michael; Uhle, Jennifer; Valentin, Andrea
Subject: Some Additional Items

Charlie, I asked my staff, particularly the staff that have been involved in responding to the Fukushima event, to put their thoughts on paper about areas they believe potentially warrant further study. My TA, Brett Rini, collected the information and I am attaching it for you and your team's consideration. Some of the items are duplicates of the ones I have already sent you, some are self-explanatory, and others just identify a general area of concern. In the interest of time, I have not attempted to edit their thoughts. If you have any questions about any of these suggestions, contact me or Brett Rini, and we can get a clarification for you.

I have also added one additional item to the list I originally sent you (item #22) and this is attached as well.

W/330

Potential Long term Issues & Research Areas as a Result of Japanese Earthquake/Tsunami and Impact on Nuclear Power Plants

Electrical Power / Station Black-out

- Assess plant response to long-term loss of onsite and offsite electrical power, as well as capabilities for mitigation (DE)
- Evaluate battery discharge duration when operated under light load (DE)
- Do we need to revisit the need for non-AC dependent hydrogen igniters on IC plants?
- Do we need AC-powered (with battery backup) hydrogen igniters in reactor buildings and/or in the vicinity of SFPs?
- Do plants have EDGs and their associated fuel tanks sufficiently protected from natural phenomena, especially floods?
- Assess the feasibility of licensees developing procedures to bring in portable electric generators to the site to a prepared location, and connecting the generators to the plant electric system. (DE)
- Assess the feasibility of developing procedures to bring in a 125 VDC battery bank and connect it to the plant DC system. (DE)
- Re-assess SBO capabilities at U.S. plants (DE)
- Should SBO coping strategies be seismically qualified to help mitigate beyond design basis seismic events where restoration of offsite power could be delayed beyond the coping time.

Instrumentation & Controls

- Do we have sufficient instrumentation in plants to accurately assess plant conditions following an accident, including severe accidents (e.g., water levels at various locations)? Is the instrumentation sufficiently robust to survive in the accident conditions?
- Is there additional instrumentation that would be of use to help manage a severe accident, such as hydrogen sensors, and would additional measures be necessary to ensure they are viable during a severe accident.
- Consider the need for additional severe accident monitoring instrumentation. Consideration should be given to providing for remote readings from the instrumentation at locations away from the unit. Wireless technology could potentially minimize the cost involved. (DE)
- Reassessment of instrumentation that can provide details on the progression of a severe accident; include remote monitoring of temperatures, pressures and radiation levels using high-capacity (long term) batteries (DE)

Reactor Pressure Vessel & Reactor Coolant System

- Performance issues of degraded/aged components: (DE)
    - Thermal loading: thermal shock, thermal transients
    - Pressure loading: explosive loadings, from thermal transients
- Components/Structures/Materials Performance in Severe or Beyond Design Basis Accidents: (DE)
    - Pumps/Valves
    - Seismic loading
- Weld Residual Stress Compendia: (DE)
    - Database of residual stresses of nuclear components: measurements & model results

- Materials research on the impact of lake/river/sea water used as makeup water to the reactor coolant system and SFP during an accident and impact on subsequent establishment of recirculation. (DE)

Containment

- PWR Containments do not have filtered vents. It is also not clear if they have vents that can be operated without AC power. Consider evaluating the benefits of putting a filtered vent on a PWR containment, along with vents that can be actuated without AC power (e.g. compressed air).
- reevaluate the need for filtered containment venting (DE)
- GSI-191 impact from seawater (DE)
- Assess coatings in the severe accident environment (DE)

Severe Accidents & Mitigation

- Effectiveness of SAMGs and EDMGs provisions (including operator training) (DRA)
- Develop SAMGs that include procedures for a containment breach (DE)
- Assess effects of high general radiation levels from a core melt on the ability for personnel to man control rooms and implement SAMGs (DE)
- Assess the need for additional regulatory guidance for severe accidents (DE)
- Review Severe Accident Management Guidelines/Emergency Operations Plans (DE)
    - Check core and spent fuel cooling procedures
        - Identify any materials issues with the cooling procedures (use of salt/river water in an emergency
- Do U.S. plants have the capability to inject ultimate heat sink water? How much time do plants with cooling ponds, like Palo Verde, have if they injected their ponds? Does that affect long term cooling strategies?
- Emergency H2 venting and whether current US plant configurations could lead to pockets of H2 in areas not covered by H2 igniters or recombiners, that give rise to explosive power sufficient to damage BWR secondary containments. (DRA) Adequacy and placement of hydrogen recombiners/igniters (DE)
- Are there accident management strategies in place for lower vessel flooding, and how well do we understand whether lower vessel flooding will work to retain a molten core inside the vessel?
- Fukushima 3 had several MOX fuel assemblies in it. How would a core with more or a full load of MOX assemblies affect the outcome of severe accidents?

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16.) There are licensees on gulf and east coast sites (e.g., Waterford) that are or may be near other industrial facilities. How well are these facilities protected against extreme environmental events, and could failures (e.g. toxic gas release, explosions of flammable liquids and gases)at these facilities due to extreme environmental events render the control room at adjacent nuclear facilities uninhabitable?

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21) For multi-unit sites, licensees are only required to mitigate the security related event at one unit under B.5.b. As a result, there may only be one piece of critical equipment to serve two or more units. Furthermore, each unit may need to carry out several strategies, such as core and spent fuel pool so the equipment may only support one strategy at a time. The

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23.) Price-Anderson – Current Price-Anderson provides about $10B+ coverage in the event of a nuclear accident. Based on what occurred at Fukushima, is the current Price-Anderson coverage still considered adequate?

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Wednesday, April 13, 2011 7:15 AM
To: Kauffman, John
Subject: RE: Draft e-mail

This is great stuff.

Even though SBO has been sent to the Task Force, I would like to forward this to Rich/Doug as additional information. The only question I have is the need for a reference (or to clearly identify it) for the EMP Attack paragraph.

Ben

From: Kauffman, John
Sent: Tuesday, April 12, 2011 10:15 AM
To: Beasley, Benjamin
Subject: Draft e-mail

Ben,

I suggest we send something like the following to the Near-Term Task Force.

One of the areas the Near-Term Task Force is charted to evaluate, based on the recent Fukushima Daiichi events, is Station Blackout.
In addition to improving NPP station blackout "coping times," we believe that minimizing the occurrence of extended duration losses of offsite power events (LOOP) (a necessary pre-condition to an extended station blackout), and enhancing the NPPs capabilities to cope with extended LOOPs are prudent. We base this conclusion on selected operating experience and information (provided below) that we have collected in the Generic Issues Program. The Operating Experience information below shows that LOOPs are typically precursor events (risk significant), and that extended duration LOOPs can result from grid collapse or severe natural events such as hurricanes, ice storms, and tornadoes; in addition to earthquakes/flooding as occurred at Fukushima Daiichi. Although the NRC does not regulate the grid, the Generic Issues Program information shows there the grid is vulnerable to Electromagnetic Pulse (EMP) attacks (act of war) and geomagnetic storms potentially causing lengthy, large loss of the grid events. Because there are numerous ways for extended LOOPs/SBOs to occur, it is important that their occurrence be minimized and that NPPs (reactors and spent fuel storage) can cope with such events if they do happen.

<u>Selected Operating Experience Documents on External Events</u>

Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20 - 30, 1992 (extended LOOP) ADAMS ML063550235

Accident Sequence Precursor (ASP) Significant Precursors
http://nrcweb.nrc.gov:8600/RES/projects/ASP/documents/Library/Significant%20Precursors/Significant%20Precursors%20(Date).pdf 4 0f 34 significant precursors involved LOOP, or partial LOOP (Items 2, 13, 21, and 28)

ASP LOOP Precursors (from FY2010 ASP SECY
http://nrcweb.nrc.gov:8600/RES/projects/ASP/documents/Library/Past%20ASP%20SECY%20Papers/SECY-10-0125.pdf) 25 LOOP ASP precursors between FY2001 and FY2009. Typically all LOOPs are ASP precursors.

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IN 92-042, Failure of Electrical Power Equipment Due to Solar Magnetic Disturbances
http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/1990/in90042.html

2003 Northeast Blackout https://reports.energy.gov/BlackoutFinal-Web.pdf
Generic Letter 2006-02 Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power
http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/2006/gl200602.pdf

Selected Information from Generic Issues Program activities
http://www.narucmeetings.org/Presentations/NARUC%20EMP%20Presentation.pdf (see page 8 for effects and page 13 for cost estimates)

EMP attack – potentially 70 to 90 % of U.S. population "unsustainable" (long term blackout, breakdown of transportation, water, energy infrastructure) Both EMP attack and electromagnetic storms can destroy large numbers of big transformers, which have limited manufacturing capacity and long-lead times (1-2 years) to replace.

Additional background information: Pre-GI-005 EMP Attack Threat
http://www.internal.nrc.gov/RES/projects/GIP/Pre-GenericIssues.html
EMP Commission Report http://www.empcommission.org/docs/A2473-EMP_Commission-7MB.pdf

Information on Geomagnetic Storms
100-year Solar Flare – potential long term blackout affecting > 130 million people
http://science.nasa.gov/science-news/science-at-nasa/2009/21jan_severespaceweather/

Bensi, Michelle

From: Bensi, Michelle
Sent: Wednesday, April 13, 2011 3:35 PM
To: Beasley, Benjamin
Subject: FW: ACTION: FAQ repository for Public Distribution

From: Ibarra, Jose
Sent: Wednesday, April 13, 2011 2:50 PM
To: Bensi, Michelle; Killian, Michelle
Subject: FW: ACTION: FAQ repository for Public Distribution

Shelby and Michelle,

I understand that OEGIB has worked on Japanese Nuclear Event Q&As. Please see OEDO effort to put all the Japanese Nuclear Event into a Share Point site. I have sent this information to John Kauffman. See me if you have any questions. Thanks. Jose

---

From: Rini, Brett
Sent: Wednesday, April 06, 2011 6:01 PM
To: Ramirez, Annie; Ibarra, Jose; Rivera-Lugo, Richard
Subject: ACTION: FAQ repository for Public Distribution

TAs,

The action on this ticket is to determine if your divisions have generated any Q&As that aren't listed at the site below. If they aren't listed, we need to compile them and send them to OPA. If they are listed, then we don't have any actions.

I would think this applies to all the divisions. Can you provide me input by next Wednesday?

Thanks,
Brett

From: Case, Michael
Sent: Wednesday, April 06, 2011 10:34 AM
To: Sheron, Brian; RidsResOd Resource; Uhle, Jennifer; Valentin, Andrea; RidsResPmdaMail Resource
Cc: Rini, Brett; Coe, Doug; Correia, Richard; Gibson, Kathy; Richards, Stuart; Scott, Michael
Subject: RE: FOR TICKETING?? FW: FAQ repository in NRR

I think they already did. I checked a couple (Indian Point seismic and did the Japanese underestimate) and they are the answers that were in Annie's Q&A set.

From: Sheron, Brian
Sent: Wednesday, April 06, 2011 10:01 AM
To: RidsResOd Resource; Uhle, Jennifer; Valentin, Andrea; RidsResPmdaMail Resource
Cc: Rini, Brett; Case, Michael; Coe, Doug; Correia, Richard; Gibson, Kathy; Richards, Stuart; Scott, Michael
Subject: RE: FOR TICKETING?? FW: FAQ repository in NRR

Please ticket to Brett. Brett, please work with Divisions on this.

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Mike, do we/can we post Annie's seismic FAQs on this site?

From: Flory, Shirley **On Behalf Of** RidsResOd Resource
Sent: Wednesday, April 06, 2011 9:18 AM
To: Sheron, Brian; Uhle, Jennifer; Valentin, Andrea; RidsResPmdaMail Resource
Subject: FOR TICKETING?? FW: FAQ repository in NRR

Brian: Should this be ticketed?

Thanks - Shirley

From: Muessle, Mary
Sent: Tuesday, April 05, 2011 6:47 PM
To: RidsNmssOd Resource; RidsResOd Resource; RidsFsmeOd Resource; RidsNroOd Resource; RidsNsirOd Resource
Cc: Schum, Constance; Pulliam, Timothy; Valentin, Andrea; Webber, Robert; Brenner, Eliot; Hayden, Elizabeth; Rothschild, Trip; Leeds, Eric; Nelson, Robert; Markley, Michael; Oesterle, Eric; Rihm, Roger; Ellmers, Glenn; Andersen, James; Landau, Mindy; Frazier, Alan; Sealing, Donna; Ficks, Ben; Holonich, Joseph; Bowman, Gregory; Rheaume, Cynthia
Subject: FAQ repository in NRR

As you may know, NRR has established a very comprehensive SharePoint site for Frequently Asked Questions regarding the Japan event. These questions were initially intended to be used internally so that all staff responding to questions from stakeholders could provide a consistent response and so that similar questions would not have to be researched several times over. The site is located at: http://portal.nrc.gov/edo/nrr/dorl/japan/Shared%20Documents/Questions%20and%20Answers.aspx

We would like to make this FAQ site available to the public as the primary consolidated site for all FAQs related to the event. To this end, I am asking your assistance by notifying us as to whether FAQs have been gathered in your office and would be appropriate for the public site. The FAQs should be sufficiently "high-level" so that they would typically be asked by a member of the public. We are not seeking very technical, detailed FAQs. They should also be FAQs that do not already appear on the SharePoint site. If your office has developed such FAQs, please send them to Beth Hayden, in OPA, who has agreed to review them to ensure they are appropriate for public release. You should then forward the OPA-approved FAQs to NRR (Eric Oesterle) for incorporation on to the SharePoint site.

Our goal is to make the site available over the course of the next week or so and then incorporate any additional OPA-vetted FAQs on to the site as soon as practicable.

Please let Mindy Landau or I know if you have any questions and thank you for your assistance and thank to NRR for this outstanding initiative!

Mary

Lee, Richard

From: Lee, Richard
Sent: Thursday, April 14, 2011 6:41 PM
To: Salay, Michael
Subject: RE: are you returning on 04/16

O.k. Have a safe trip.

From: Salay, Michael
Sent: Thursday, April 14, 2011 5:01 PM
To: Lee, Richard
Subject: RE: are you returning on 04/16

Yes. I haven't heard otherwise.

-Mike

From: Lee, Richard
Sent: Thursday, April 14, 2011 4:56 PM
To: Salay, Michael
Subject: are you returning on 04/16

Mikey-san:

Are you returning on 4/16?

Richard

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Beasley, Benjamin

From: Beasley, Benjamin
Sent: Monday, April 18, 2011 11:31 AM
To: Kauffman, John
Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs
Attachments: ATT00001..gif; ATT00002..gif

Your thoughts?

From: Sheron, Brian
Sent: Monday, April 18, 2011 11:20 AM
To: Beasley, Benjamin
Cc: Correia, Richard; Coe, Doug
Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

See below. Would this likely pass a cost-benefit backfit test?

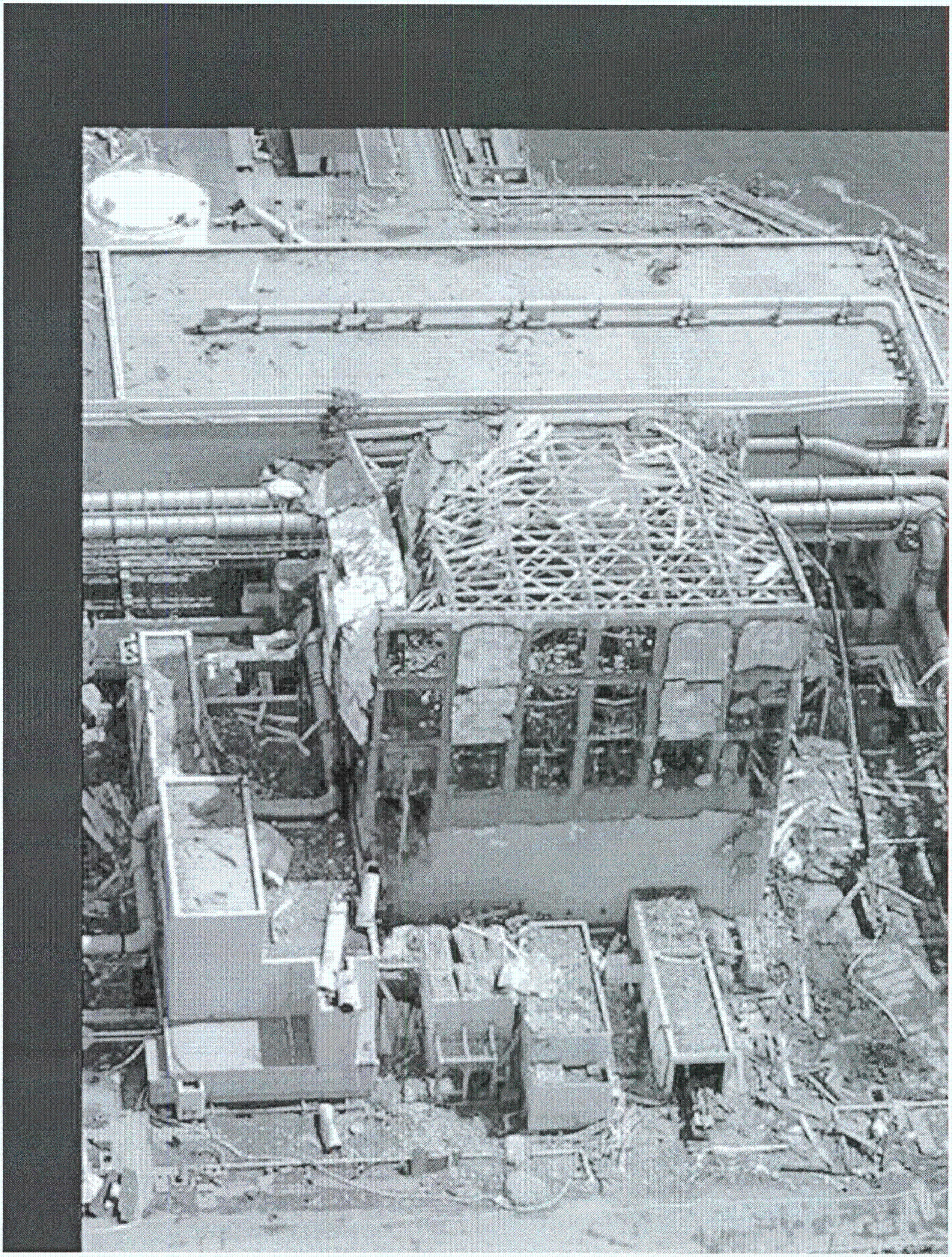
From: Richard L Garwin [mailto:rlg2@us.ibm.com]
Sent: Sunday, April 17, 2011 4:25 PM
To: Larzelere, Alex
Cc: Caponiti, Alice; Busby, Jeremy T; DL-NITsolutions; Schneider, Steve
Subject: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Dear Colleagues,

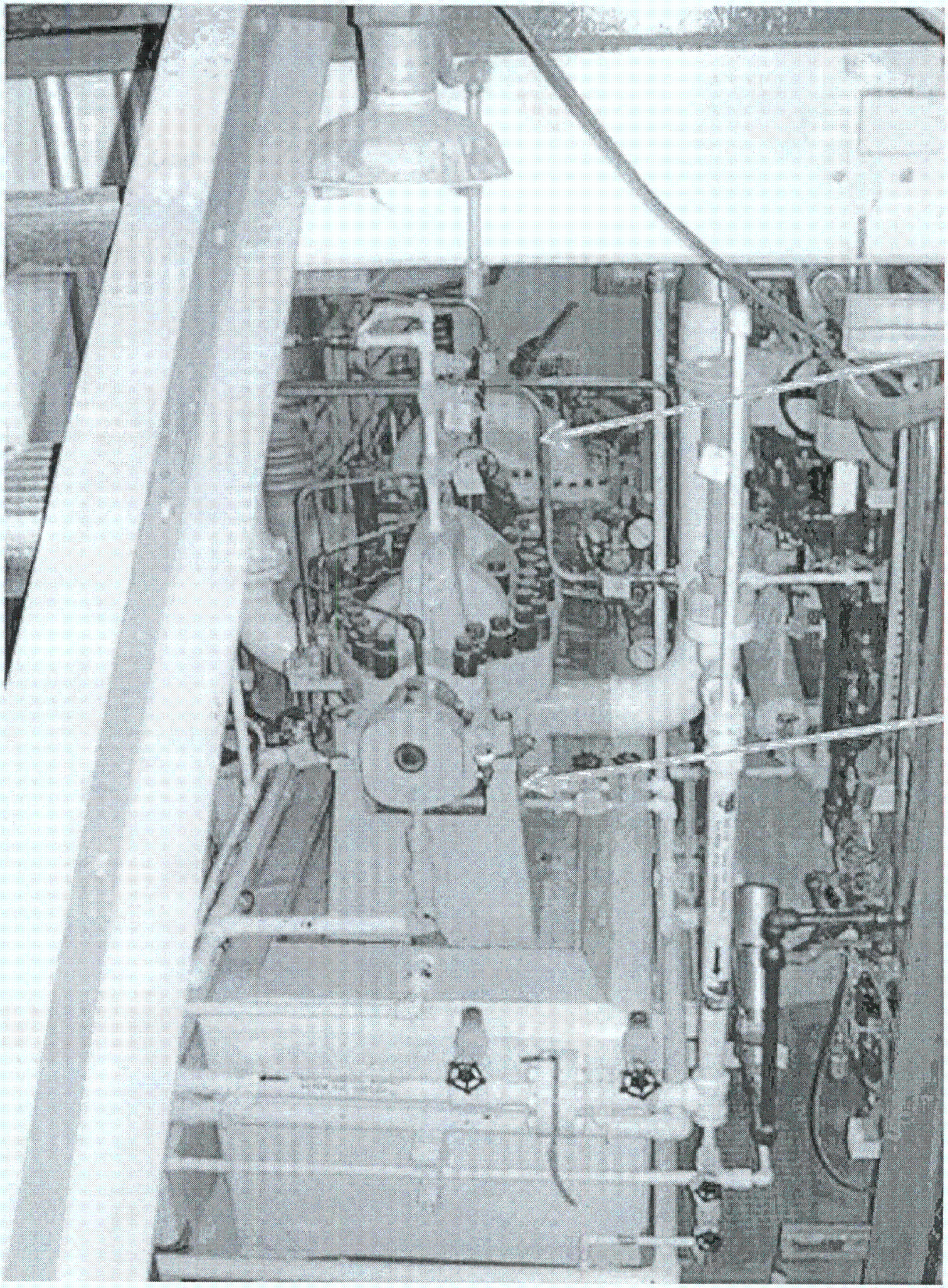
http://allthingsnuclear.org of April 14 has a very useful presentation of the Fukushima Dai-ichi problem.

I attach the first slide and also a detail of the steam-driven "isolation turbine and pump," and provide also
a SUGGESTION by Bill Press.

W334



Reactor Core Isolation Co



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Dick Garwin

Beasley, Benjamin

From: Beasley, Benjamin
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Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs
Attachments: ATT00001..gif; ATT00002..gif

Rich and Doug,

John Kauffman and I discussed this idea today. We will have a proposed response for your comment tomorrow or Wednesday.

Ben

From: Sheron, Brian
Sent: Monday, April 18, 2011 11:20 AM
To: Beasley, Benjamin
Cc: Correia, Richard; Coe, Doug
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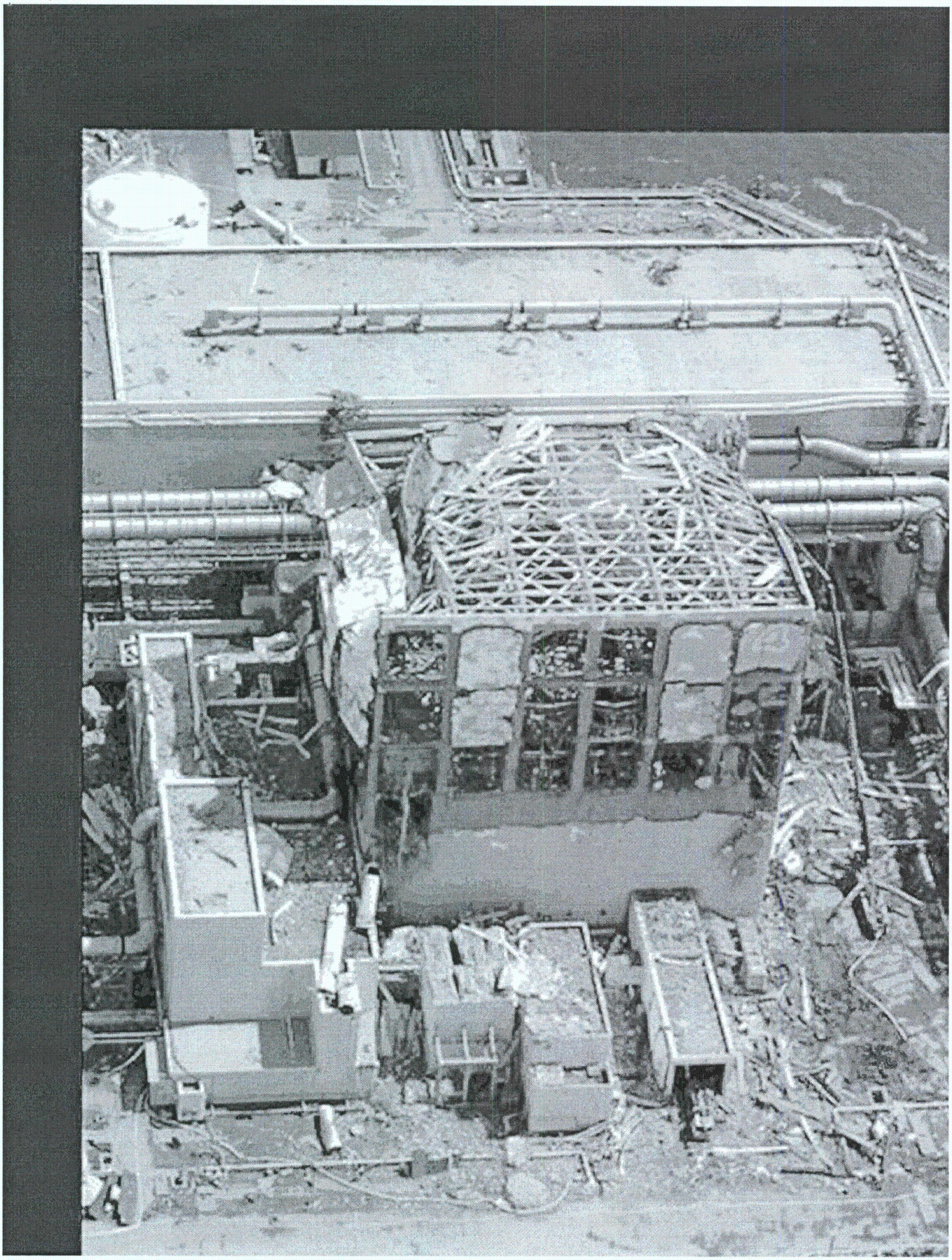
4/335



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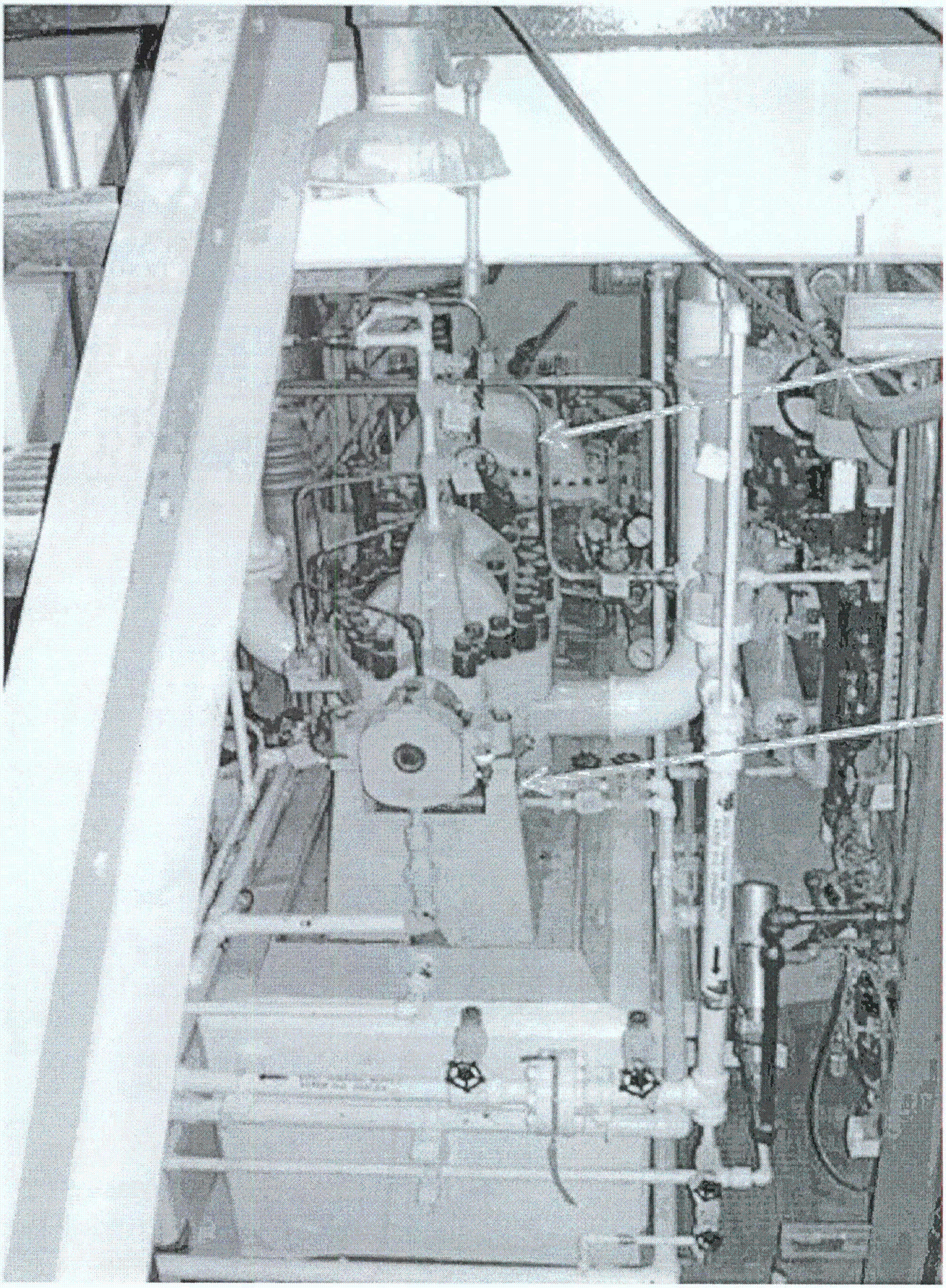

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Dick Garwin

Beasley, Benjamin

From: Beasley, Benjamin
Sent: Tuesday, April 19, 2011 5:12 PM
To: Correia, Richard; Coe, Doug
Cc: Kauffman, John
Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs
Attachments: image001.gif; image002.gif

Rich and Doug,

Below are our thoughts on the RCIC generator idea. Feel free to edit and forward this to Brian when you are ready.

Ben

From: Kauffman, John
Sent: Tuesday, April 19, 2011 2:59 PM
To: Beasley, Benjamin
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Ben,

As we discussed, this is an interesting idea...good outside the box thinking! That said, I think the answer to Brian's question is probably "no," as discussed below. You may want to have Marty or Gary check my PRA numbers/logic. Plants might voluntarily want to develop ways to use RCS energy and decay heat to generate electrical power in a long-term, "non-recoverable SBO," rather than just being helpless.

Risk Discussion

From the latest update to NUREG/CR-5750, the initiating event frequency for LOOP has an industry mean of 3.5E-02 per year. From the latest update to NUREG/CR-5500, Vol. 5, the industry-wide average failure probability for not completing the 8-hour emergency power system (EPS) mission time is 9E-04 per year.

So the probability of a LOOP followed by failure of the onsite EPS roughly 3E-5 per year. Typically, under the agency's Regulatory Analysis Guidelines (page 14), an item cannot pass backfit if the estimated risk reduction in CDF is less than 1E-05 per year. The LOOP IE frequency and EPS 8-hour failure probability do not credit grid recovery actions, or alternate AC capabilities (SBO or B5B diesels). When expecting severe weather, some plants shutdown and pre-position skid-mounted EDGs. Therefore, the CDF due to station blackout is probably less than 1 E-05.

That said, as evinced by the recent Japanese earthquake and tsunami, the grid and EPS are vulnerable to common cause failures due to extreme external events (most notably seismic and flooding)

In summary; our sense is that any backfits in this area will need to use an adequate protection justification.

Systems Discussion

RCIC is a small system (typically 400 gpm (larger on higher MW plants))...don't know how much electricity it could generate.

In a LOOP event, RCIC typically runs (along with HPCI/HPCS to restore/maintain reactor water level). However, these systems have more capacity than is needed and either trip on high level or require operator

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intervention to throttle them back. The point is that RCIC only runs intermittently. It also does not run at constant speed, which would be problematic for making stable, useable AC. (It could be useful for keeping batteries charged.)

Connecting a magneto to RCIC would create a "load," so it seems that RCIC would either need to draw more steam to produce the same injection flow or be de-rated. If the RCIC turbine were run continuously, it could depressurize the RCS, causing a loss of motive force. A separate turbine or a generator connected to the RCIC turbine only when RCIC is not injecting would address the de-rating issue but not the depressurization issue. Either of these would likely be more costly than alternatives.

In summary; our sense is that this idea would present challenging hurdles to implement. A more straight-forward approach would be to have pre-arranged ways to connect temporary (pre-arranged) AC sources, e.g. skid mounted EDGs, to the station's emergency/vital buses.

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Sent: Monday, April 18, 2011 11:20 AM
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Cc: Correia, Richard; Coe, Doug
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See below. Would this likely pass a cost-benefit backfit test?

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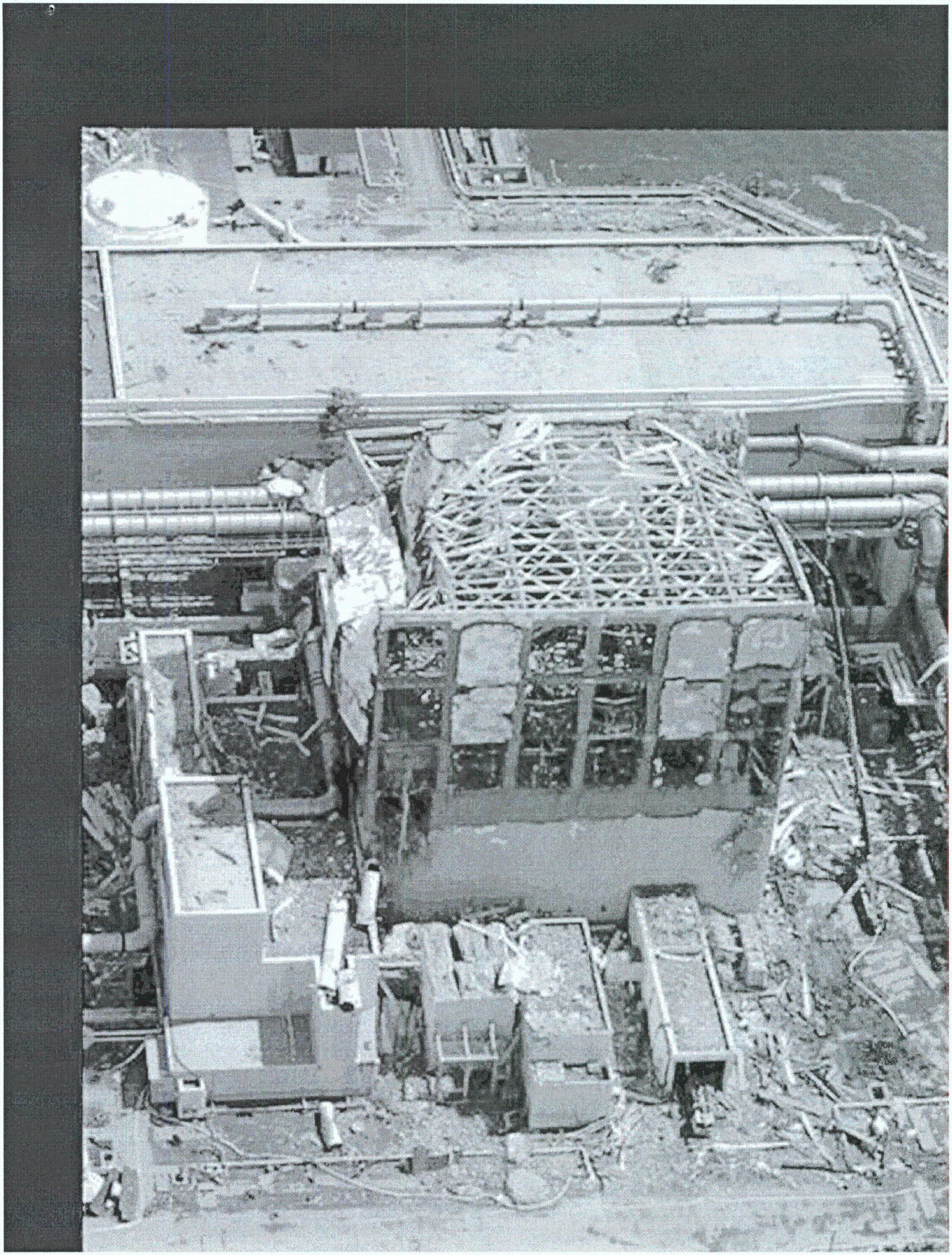

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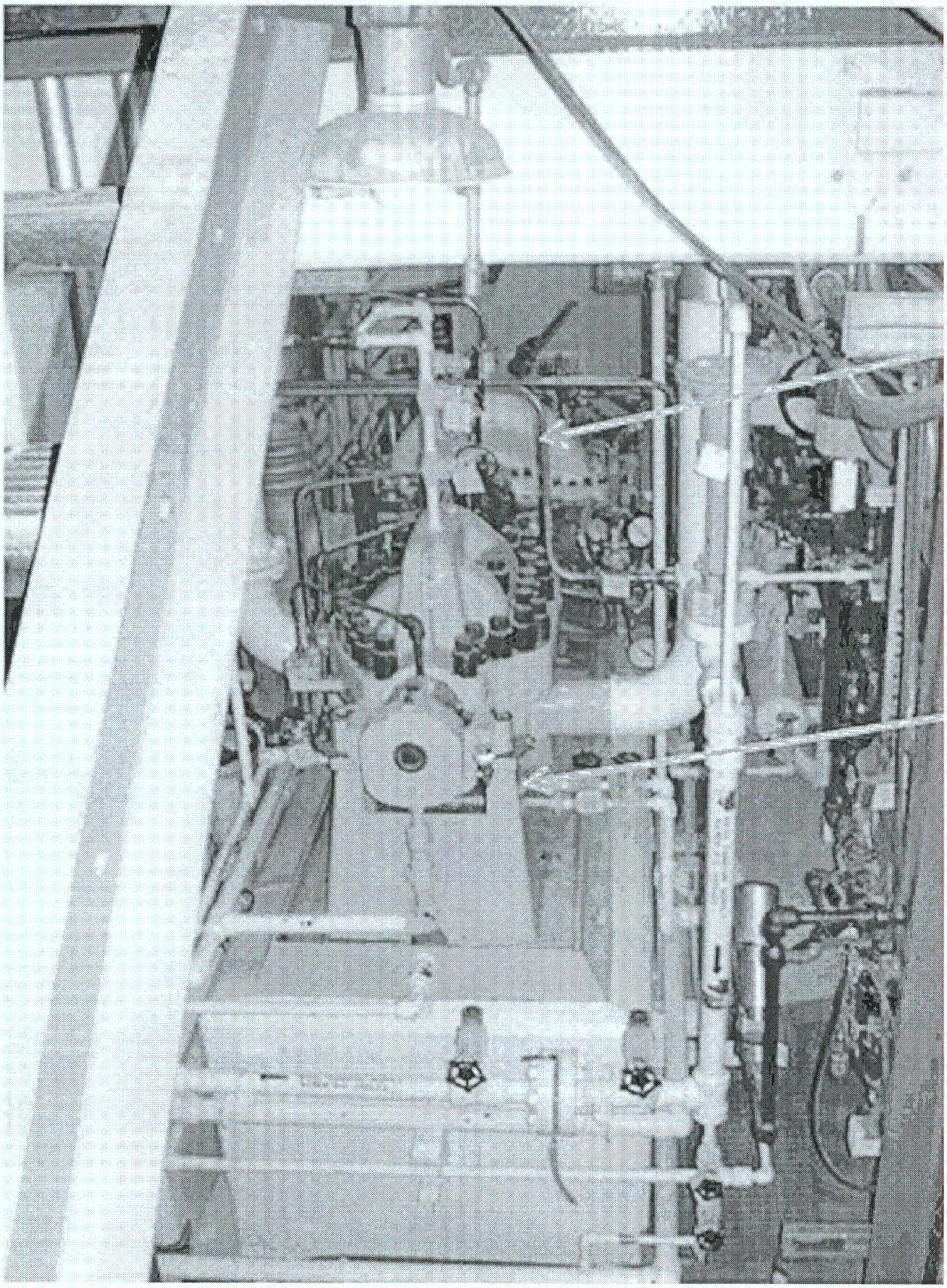

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Dick Garwin

Huffert, Anthony

From: Huffert, Anthony
Sent: Wednesday, April 20, 2011 2:32 AM
To: Meighan, Sean; Gepford, Heather; Conatser, Richard
Cc: Huffert, Anthony
Subject: "Manual for Measuring Radioactivity of Foods in Case of Emergency" dated May 9, 2002

H.ere is the link to the "Manual for Measuring Radioactivity of Foods in Case of Emergency" dated May 9, 2002
http://www.mhlw.go.jp/stf/houdou/2r9852000001558e-img/2r98520000015cf6.pdf
http://www.mhlw.go.jp/stf/houdou/2r9852000001558e-img/2r98520000015cfn.pdf

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Huffert, Anthony

From: Huffert, Anthony
Sent: Wednesday, April 20, 2011 4:51 AM
To: Gepford, Heather; Meighan, Sean
Subject: REMINDER: Questions from PMT for task

From: Hoc, PMT12
Sent: Thursday, April 14, 2011 1:42 PM
To: RST01 Hoc; Huffert, Anthony
Cc: Hart, Michelle; Watson, Bruce; OST01 HOC
Subject: Questions from PMT for task

Hello RST and Japan Team,

Can you please respond to these two questions to assist the staff in answering a question from NARAC on the development of new source terms. Please respond to all. NRC staff points of contact are Bruce Watson and Michelle Hart. This is not an action, just a question – we are just looking for information.

- Do we have anyone recreating the source term from the reactors and SFP based on plant conditions or field measurement readings?
- Are there updates on releases or degree of core damage based on plant data?

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Beasley, Benjamin

From: Beasley, Benjamin
Sent: Wednesday, April 20, 2011 7:35 AM
To: Correia, Richard; Coe, Doug
Cc: Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs
Attachments: image001.gif; image002.gif

I am satisfied. John used the same sources that Gary would. We could quibble over using the mean versus the 95% number, but the difference is small.

Ben

From: Correia, Richard
Sent: Wednesday, April 20, 2011 7:07 AM
To: Beasley, Benjamin; Coe, Doug
Cc: Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Ben,

John asked to have Marty or Gary check his PRA logic/numbers. Are you satisfied they are accurate? John discussion and system performance seems reasonable to me.

thx

Richard Correia, PE
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
US NRC

richard.correia@nrc.gov

From: Beasley, Benjamin
Sent: Tuesday, April 19, 2011 5:12 PM
To: Correia, Richard; Coe, Doug
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4/339

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Cc: Correia, Richard; Coe, Doug

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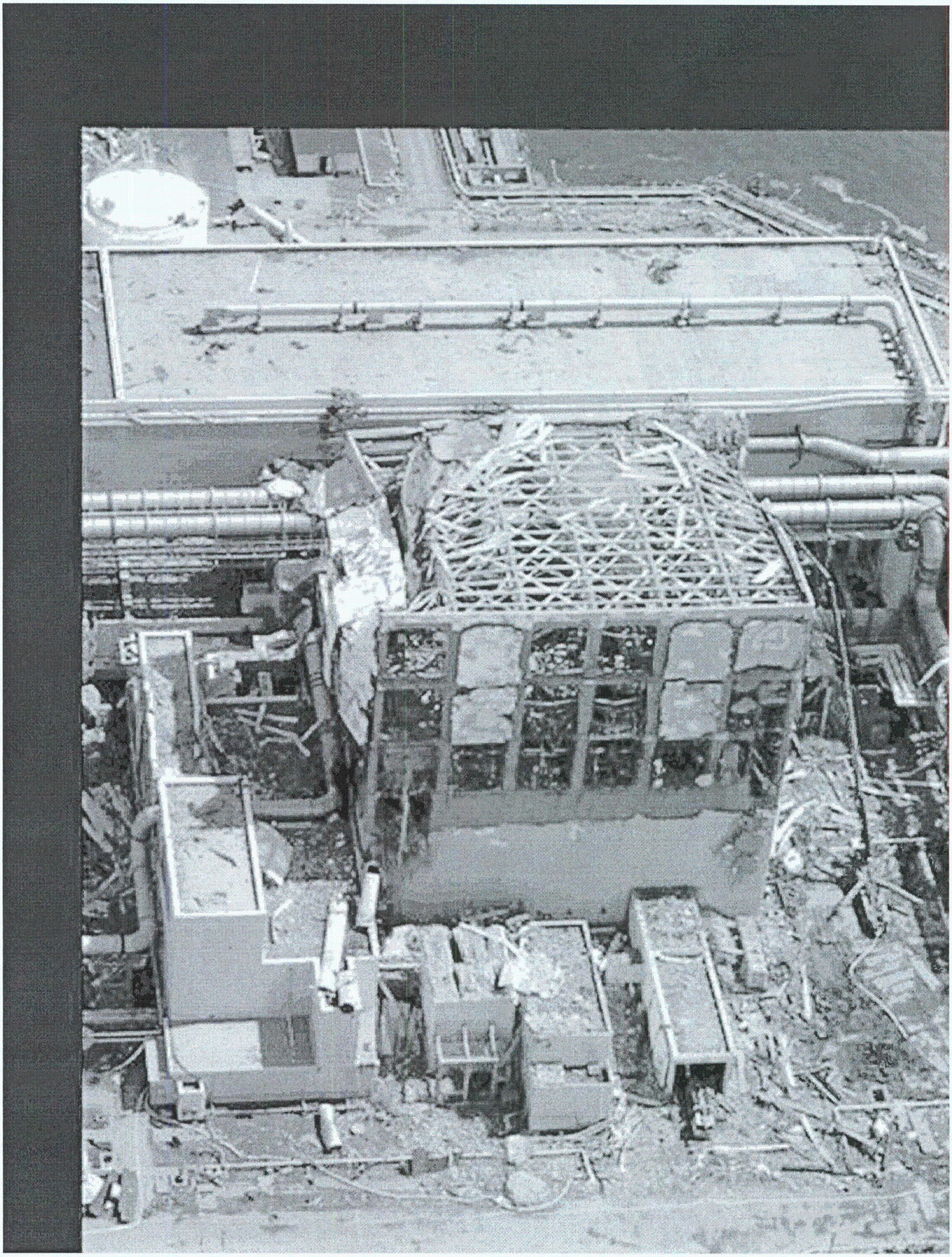

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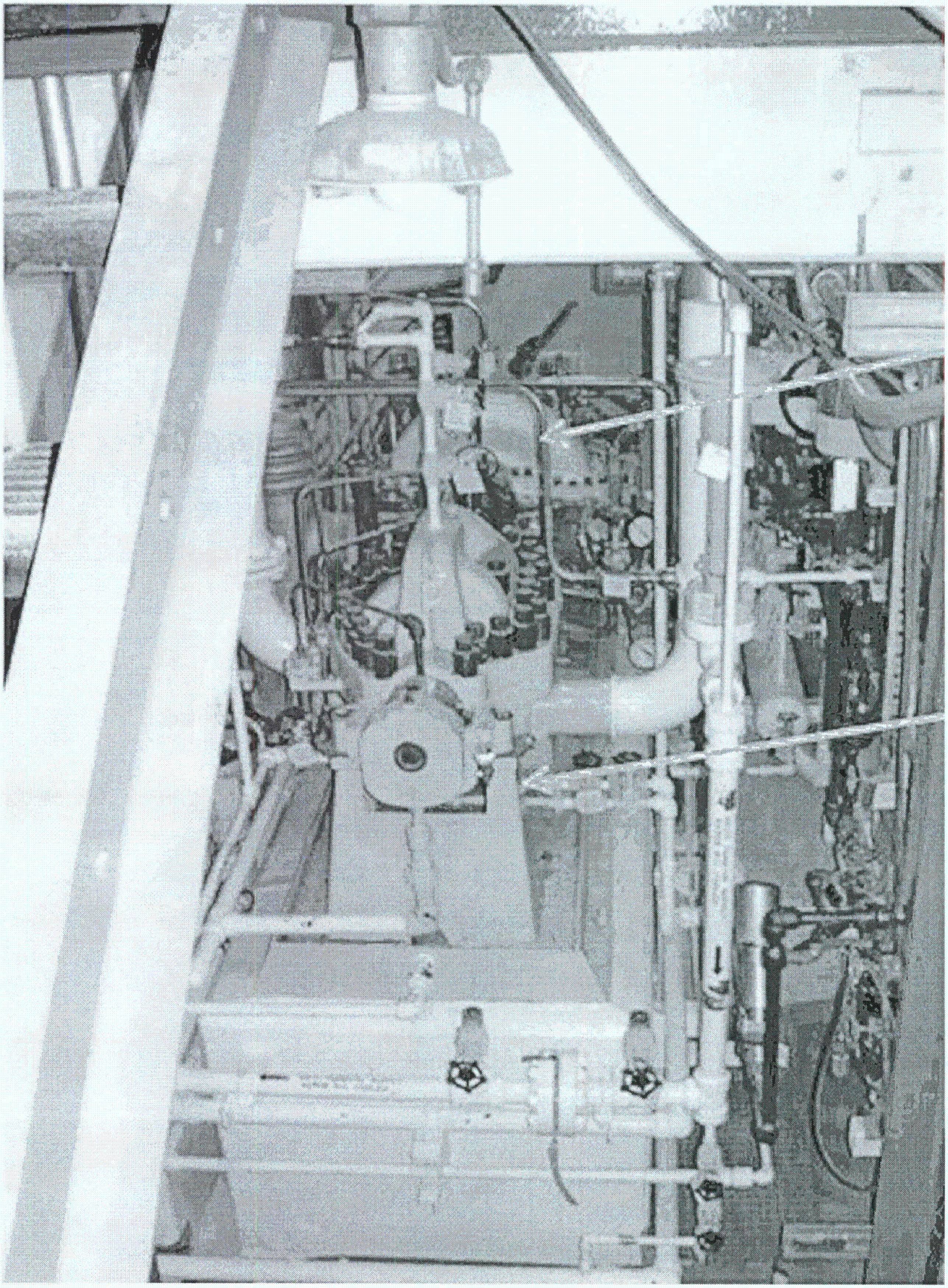

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Beasley, Benjamin

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To: Coe, Doug; Correia, Richard; Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs
Attachments: image001.gif; image002.gif; image003.png

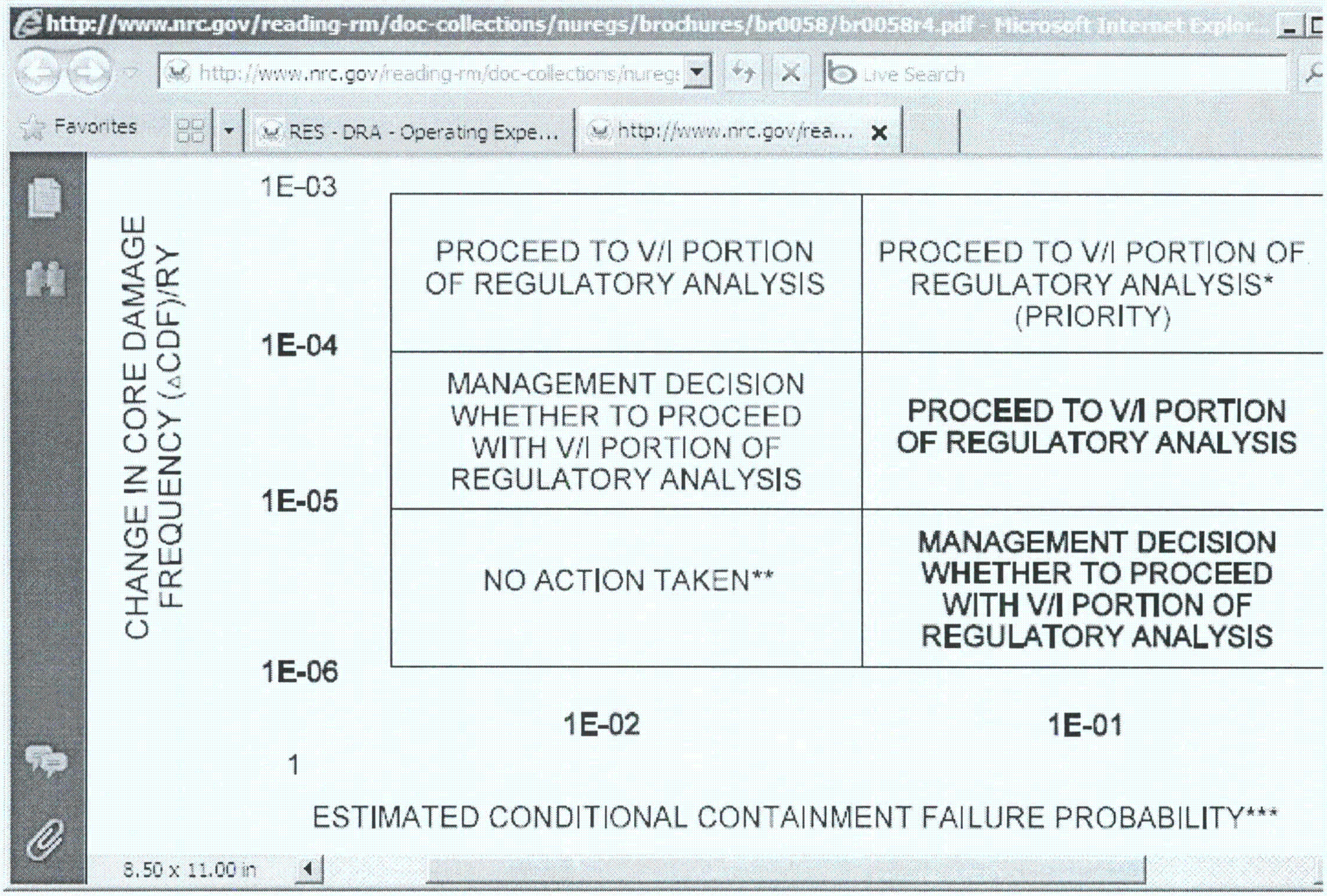
Doug,

You are correct that consequences are used to establish the cost-benefit. But the guidance says that the item can be dismissed if the risk numbers are below a certain level. In the chart below, for BWR Mark1 plants, we would use a conditional containment failure probability of E-1. With the delta CDF below E-5, we are in the zone of "Management decision whether to proceed." If we consider other things that may have driven the risk numbers down, like B5B, etc., then we could be below E-6 and in the "No action" zone altogether.

I agree with you regarding zeroing in on particular solutions. There would be much to consider before pursuing a solution like this. We were responding under the assumption that Brian was asking about regulatory feasibility, but we felt it worth pointing out some of the system implications.

We will consolidate our thoughts and send a response to Brian.

Ben



4340

From: Coe, Doug
Sent: Wednesday, April 20, 2011 8:30 AM
To: Beasley, Benjamin; Correia, Richard; Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Ben/Rich/John,
Two reactions:

1. My initial reaction to Brian's question was: *you need to do the analysis* and shouldn't SWAG this. As I read the discussion below, it is focused on core damage, whereas the reg analysis must consider consequences (i.e. person-rem avoided). The core damage piece doesn't apparently consider SBO coping equipment/procedures (I'm not sure why) and the reg analysis guidelines do not (I believe) address multi-unit severe external events. So.... back to 'you shouldn't SWAG this.'
2. Second, I would resist zeroing in on specific 'solutions' without a full and integrated review of how any/all 'solutions' would impact the overall reactor plant system and its risk profile. Adding any new backfit carries the potential for creating new vulnerabilities even as you are attempting to resolve known vulnerabilities. I would advocate continuing to collect ideas such as this one, but not to do any 'cost-benefit' or similar analysis until we can look at them in an integrated manner.

From: Beasley, Benjamin
Sent: Wednesday, April 20, 2011 7:35 AM
To: Correia, Richard; Coe, Doug
Cc: Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

I am satisfied. John used the same sources that Gary would. We could quibble over using the mean versus the 95% number, but the difference is small.

Ben

From: Correia, Richard
Sent: Wednesday, April 20, 2011 7:07 AM
To: Beasley, Benjamin; Coe, Doug
Cc: Kauffman, John
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Ben,

John asked to have Marty or Gary check his PRA logic/numbers. Are you satisfied they are accurate? John discussion and system performance seems reasonable to me.

thx

Richard Correia, PE
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
US NRC

richard.correia@nrc.gov

From: Beasley, Benjamin
Sent: Tuesday, April 19, 2011 5:12 PM
To: Correia, Richard; Coe, Doug
Cc: Kauffman, John
Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Rich and Doug,

Below are our thoughts on the RCIC generator idea. Feel free to edit and forward this to Brian when you are ready.

Ben

From: Kauffman, John
Sent: Tuesday, April 19, 2011 2:59 PM
To: Beasley, Benjamin
Subject: RE: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

Ben,

As we discussed, this is an interesting idea...good outside the box thinking! That said, I think the answer to Brian's question is probably "no," as discussed below. You may want to have Marty or Gary check my PRA numbers/logic. Plants might voluntarily want to develop ways to use RCS energy and decay heat to generate electrical power in a long-term, "non-recoverable SBO," rather than just being helpless.

Risk Discussion
From the latest update to NUREG/CR-5750, the initiating event frequency for LOOP has an industry mean of 3.5E-02 per year. From the latest update to NUREG/CR-5500, Vol. 5, the industry-wide average failure probability for not completing the 8-hour emergency power system (EPS) mission time is 9E-04 per year.

So the probability of a LOOP followed by failure of the onsite EPS roughly 3E-5 per year. Typically, under the agency's Regulatory Analysis Guidelines (page 14), an item cannot pass backfit if the estimated risk reduction in CDF is less than 1E-05 per year. The LOOP IE frequency and EPS 8-hour failure probability do not credit grid recovery actions, or alternate AC capabilities (SBO or B5B diesels). When expecting severe weather, some plants shutdown and pre-position skid-mounted EDGs. Therefore, the CDF due to station blackout is probably less than 1 E-05.

That said, as evinced by the recent Japanese earthquake and tsunami, the grid and EPS are vulnerable to common cause failures due to extreme external events (most notably seismic and flooding)

In summary; our sense is that any backfits in this area will need to use an adequate protection justification.

Systems Discussion

RCIC is a small system (typically 400 gpm (larger on higher MW plants))...don't know how much electricity it could generate.

In a LOOP event, RCIC typically runs (along with HPCI/HPCS to restore/maintain reactor water level). However, these systems have more capacity than is needed and either trip on high level or require operator intervention to throttle them back. The point is that RCIC only runs intermittently. It also does not run at constant speed, which would be problematic for making stable, useable AC. (It could be useful for keeping batteries charged.)

Connecting a magneto to RCIC would create a "load," so it seems that RCIC would either need to draw more steam to produce the same injection flow or be de-rated. If the RCIC turbine were run continuously, it could depressurize the RCS, causing a loss of motive force. A separate turbine or a generator connected to the RCIC turbine only when RCIC is not injecting would address the de-rating issue but not the depressurization issue. Either of these would likely be more costly than alternatives.

In summary; our sense is that this idea would present challenging hurdles to implement. A more straight-forward approach would be to have pre-arranged ways to connect temporary (pre-arranged) AC sources, e.g. skid mounted EDGs, to the station's emergency/vital buses.

From: Sheron, Brian
Sent: Monday, April 18, 2011 11:20 AM
To: Beasley, Benjamin
Cc: Correia, Richard; Coe, Doug
Subject: FW: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

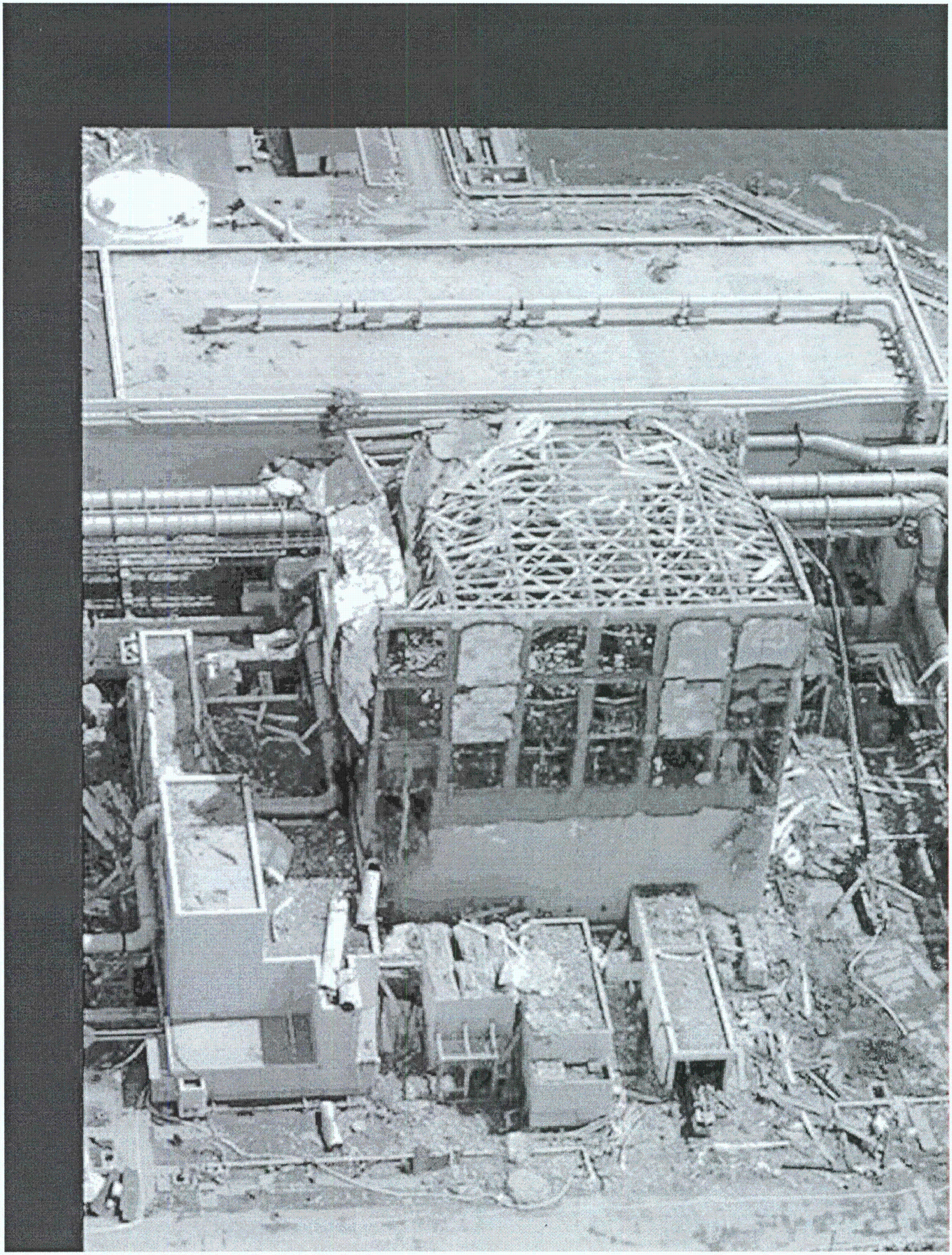
See below. Would this likely pass a cost-benefit backfit test?

From: Richard L Garwin [mailto:rlg2@us.ibm.com]
Sent: Sunday, April 17, 2011 4:25 PM
To: Larzelere, Alex
Cc: Caponiti, Alice; Busby, Jeremy T; DL-NITsolutions; Schneider, Steve
Subject: Useful presentation from http://allthingsnuclear.org of April 14, and a SUGGESTION for improving our BWRs

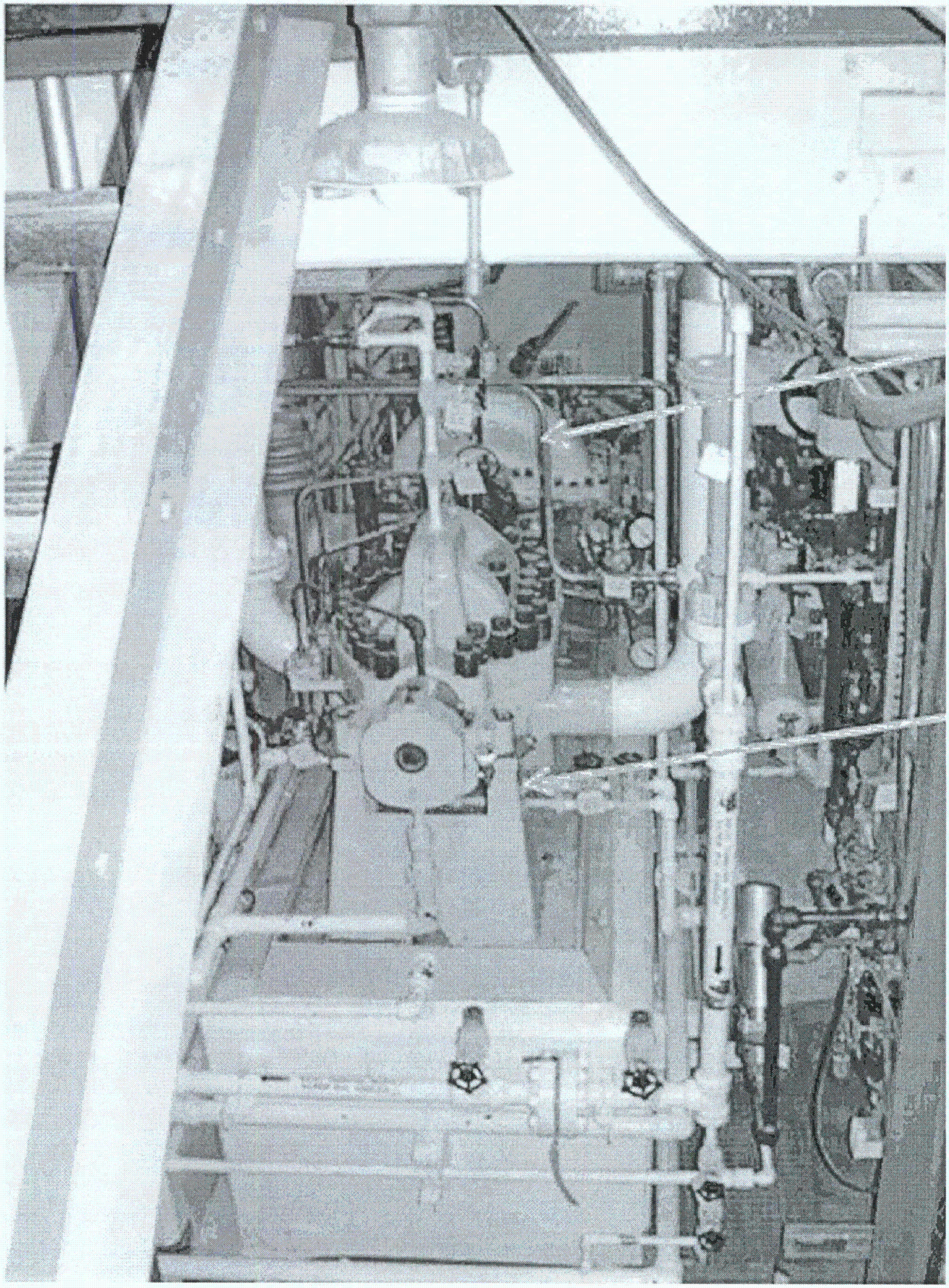
Dear Colleagues,

http://allthingsnuclear.org of April 14 has a very useful presentation of the Fukushima Dai-ichi problem.

I attach the first slide and also a detail of the steam-driven "isolation turbine and pump," and provide also
a SUGGESTION by Bill Press.



Reactor Core Isolation Co



Bill Press (William H. Press, University of Texas at Austin, and LANL) asks why the RCIC turbine/pump does not have a "magneto" on the shaft, like that on a piston-driven aircraft engine, so that whenever the pump is running there is electrical power generated for the RCIC valves and other emergency loads. This might well be used to charge the batteries, too, and operate the control room indicators and lights.

This seems to me an eminently practical suggestion, which I am passing on for communication to NE and NRC.

Dick Garwin