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

May 23, 2012

Dr. J. Sam Armijo, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, DATED APRIL 25, 2012, ON THE SPENT FUEL POOL SCOPING STUDY

Dear Dr. Armijo:

I am responding to your letter of April 25, 2012, in which you provided the comments of the Advisory Committee on Reactor Safeguards (ACRS) on the staff's Spent Fuel Pool Scoping Study that was presented to the ACRS on April 12–14, 2012.

The staff agrees with ACRS' summary of the limitations of the study. The study results and the limitations will be considered with other factors when the staff evaluates the Fukushima Lessons Learned Tier 3 issue of whether spent fuel should be transferred to dry cask storage earlier than currently planned by the nuclear power plant licensees. As summarized in your letter, such factors include, but are not limited to, the risks of additional fuel handling when loading and transferring the spent fuel to the casks. The staff will also consider the relevant operating experience related to the integrity of spent fuel pools subjected to severe seismic events.

We appreciate the comments on the staff's plans for the study and look forward to further interactions on this topic.

Sincerely,

Michael F. Weber

for R. W. Borchardt
Executive Director
for Operations

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

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## Wong, Emma

**From:** Campbell, Debbie [dcampbell@epri.com]
**Sent:** Wednesday, October 26, 2011 2:08 PM
**To:** Ahn, Tae; Alejana, Consuelo; Almoguera, Ramon; Alonso, Jose Manuel; Alsaed, Halim; Asano, Ryoji; Askarieh, Mehdi; Auzoux, Quentin; Bader, Sven; Baker, Steve; Ballinger, Ron; Barnabas, Istvan; Bateman, Mark; Behravesh, Mohamad; Bellamy, Steve; Bennett, John; Bernard, Felix; Bevilacqua, Arturo; Billone, Mike; Birk, Sandra; Bracey, Bill; Brookmire, Tom; Bunt, Randy; Buschmann, Nancy; Cairns, Martin; Carlsen, Brett; Carter, Joe; Caseres, Leonardo; Choi, Jongwon; Coaster, Don; Codee, Hans; Cole, Kent; Conde, Jose-Manuel; Connell, James; Couplet, Damien; Cummings, Kris; Danker, Bill; Danner, Tom; Darby, Sam; Davis, Demitrius; Dawson, Chris; Delannay, Michel; Di Gasbarro, Fernanda; Dobson, Alan; Duncan, Andy; Dunn, Darrell; Dyck, Peter; Easton, Earl; Edsinger, Kurt; Edwards, Steve; Einziger, Robert; Engelvaart, Marco; England, Jeff; Erhard, Anton; Fernandez, F. Javier; Fernandez, Rene; Francia, Lorenzo; Fujimoto, Takeshi; Fujita, Hirofumi; Fujiwara, Shusuke; Gago, Jose; Garamszeghy, Miklos (Mike); Garrido, David; Geiser, Heinz; Gonzalez, Hipolito; Gonzalez, Rafael; Gordon, Matthew; Gran, Per; Graves, Herman; Guimaraes, Maria; Guttmann, Jack; Haddad, Roberto; Hanson, Brady; Heck, Matthias; Herrera Nevarro, Jose-Antonio; Hinojosa, Luis; Hodgson, Zara; Honjin, Masao; Hopf, Jim; Huang, Yuhao; Hueggenberg, Roland; Ishiko, Daiichi; Issard, Herve; Iyer, Natraj; James, Richard; Jubin, Bob; Kadak, Andrew; Kapoor, Ashok; Katayama, Jiro; Kato, Masami; Kessler, John; Kitamura, Takafumi; Kook, Donghak; Kowalewski, Ron; Kumano, Yumiko; Kunerth, Dennis; Lavara, Arturo; Leblang, Suzanne; Leduc, Dan; Lesec, Valerie; Levin, Adam; Lloret, Miriam; Lombardi, Gianrico; Lopez, Jose Vicente; Machiels, Albert; MacRae, Walter; Malesys, Pierre; Marschman, Steve; Martin, Zita; McCullum, Rod; McMahon, Kevin; Metlay, Daniel; Mooney, Mike; Mote, Nigel; Murphy, William; Murray, Paul; Muthu, Nathan; Nakagome, Yoshihiro; Narayanan, Prakash; Negrini, Teresa ; Neider, Tara; Nesbit, Steve; Nichol, Mrcus; Niyogi, Kalyan; Norwood, Keith; O'Connor, Steve; Ordaz, Vonna; Ordogh, Miklos; Padovani, Cristiano; Parks, Cecil; Partes, Bettina; Pechera, Iurii; Pfeifer, Holger; Quecedo, Manuel; Resident Researcher - Jung, Dae-IL; Rigby, Doug; Robertshaw, Julian; Robins, Randy; Rubenstone, James; Saegusa, Toshiari; Sanjurjo, Manuel Novo; Sataar, Haaroen; Schroeder, Jens; Schwartz, Glenn; Seaman, Craig; Selby, Greg; Sen, Anindya; Sheffield, Richard; Shih, Peter; Shirai, Koji; Sindelar, Bob; Soares, Luiz; Soltis, Jeff; Sorenson, Ken; Sperry, Lee; Stockinger, Siegfried; Stockman, Christine; Swift, Peter; Takahashi, Tadayoshi; Tedesco, Giancarlo; Tripputi, Ivo; Twala, Vusi; van der Lee, Jan; Verhoef, Ewoud V.; Versaci, Raul; Vinson, Dennis; Voelzke, Holger; Wagner, John; Waldrop, Keith; Wall, James; Wang, Lumin; Wataru, Masami; Waters, Michael; Weiner, Ruth; Williams, Jeff; Wilson, Ian; Wolff, Dietmar; Wong, Emma; Wood, Peter; Yamamoto, Masahiro; Yamamoto, Tomofumi; Zigh, Ghani; Zuloaga, Pablo; Zurro, Julio Blanco
**Cc:** Kessler, John
**Subject:** Extended Storage Collaboration Program (ESCP): Progress Report and Review of Gap Analyses

Sent on behalf of John Kessler…

Notification of a publicly available EPRI report on extended storage:
http://my.epri.com/portal/server.pt?Abstract_id=000000000001022914

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## Wong, Emma

**From:** Kessler, John [JKESSLER@epri.com]
**Sent:** Tuesday, November 22, 2011 12:25 PM
**To:** Bracey, Bill; Carlsen, Brett; Danner, Tom; Edwards, Steve; Einziger, Robert; Hanson, Brady; Hinojosa, Luis; Kessler, John; Robins, Randy; Sisley, Steve; Sorenson, Ken; Sowder, Andrew; Weiner, Ruth; Williams, Jeff; Wong, Emma; Birk, Sandra; Bunt, Randy; Buschman, Nancy; Hopf, Jim; Kokajko, Lawrence; McCullum, Rod; Neider, Tara; Niyogi, Kalyan; Ordaz, Vonna; Saegusa, Toshiari; Schwab, Pat; Seaman, Craig; Shirai, Koji; Voelzke, Holger; Waters, Michael
**Subject:** Used Fuel Extended Storage reports of interest

Dear ESCP members:

Attached are links to two reports that may be of interest to you.

http://my.epri.com/portal/server.pt?Abstract_id=000000000001021057
http://pbadupws.nrc.gov/docs/ML1132/ML11321A182.pdf

For those of you who plan to attend the ESCP meeting here in Charlotte next month and have not yet registered online, please do so. If you have not received an electronic invitation and would like to attend, please let me know.

Sincerely,

John Kessler

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## Wong, Emma

**From:** Campbell, Debbie [dcampbell@epri.com]
**Sent:** Friday, March 09, 2012 9:20 AM
**To:** Ahn, Tae; Alejano, Consuelo; Almoguera, Ramon; Alonso, Jose Manuel; Alsaed, Halim; Asano, Ryoji; Askarieh, Mehdi; Auzoux, Quentin; Bader, Sven; Baker, Steven; Ballinger, Ron; Barnabas, Istvan; Bateman, Mark; Behravesh, Mohamad; Bellamy, Steve; Bennett, John; Bernard, Felix; Bevilacqua, Arturo; Billone, Michael; Birk, Sandra; Bracey, William "Bill"; Brookmire, Tom; Brown, James; Bunt, Randy; Buschmann, Nancy; Cairns, Martin; Cannell, Gary; Carlsen, Brett; Carter, Joe; Caseres, Leonardo; Channell, Clay; Cheng, Shih-Chung; Choi, Jongwon; Chung, Sunghwan; Coaster, Don; Codee, Hans; Cole, Kent; Conde, Jose-Manuel; Connell, James; Contractor - Greer, Bruce; Couplet, Damien; Cummings, Kris; Danker, William; Danner, Thomas; Darby, Sam; Davis, Demitrius; Dawson, Chris; Deboi, Kristi; Delannay, Michel; Di Gasbarro, Fernanda; Dobson, Alan; Duncan, Andrew; Dunn, Darrell; Dyck, Peter; Easton, Earl; Edsinger, Kurt; Edwards, Steve; Einziger, Robert; Elwood, Randy; Engelvaart, Marco; England, Jeffery; Erhard, Anton; Farnum, Cathy Ottinger; Fernandez, Rene; Floyd, Mike; Francia, Lorenzo; Fujimoto, Takeshi; Fujita, Hirofumi; Fujiwara, Shusuke; Gago, Jose; Garamszeghy, Miklos (Mike); Geiser, Heinz; Gonzalez, Hipolito; Gonzalez, Rafael; Gordon, Matthew; Gran, Per; Grant, Glenn; Graves, Herman; Grizzi, Robert; Guimaraes, Maria; Gustems, Brian; Guttmann, Jack; Haddad, Roberto; Hanson, Brady; Heck, Matthias; Herrera Nevarro, Jose-Antonio; Hinojosa, Luis; Hodgson, Zara; Hollinger, Gary; Honjin, Masao; Hopf, Jim; Howard, Robert; Huang, Yuhao; Hueggenberg, Roland; Ishiko, Daiichi; Issard, Herve; Jacobs, Christian; James, Richard; Johnson, Lawrence; Jorgensen, Vern; Jubin, Bob; Jung, Dae-Il; Jung, Hundal "Andy"; Kadak, Andrew; Kapoor, Ashok; Katayama, Jiro; Kato, Masami; Kessler, John; Kitamura, Takafumi; Kokajko, Lawrence; Kook, Donghak; Kot, Christian; Kowalewski, Ron; Kuba, Stanislav; Kumano, Yumiko; Kunerth, Dennis; Kuo, Roang-Ching; Lambert, John; Lavara, Arturo; Lavender, Curt; Leblang, Suzanne; Leduc, Daniel; Lesec, Valerie; Levin, Adam; Liu, Yung; Lloret, Miriam; Lombardi, Gianrico; Lopez, F. Javier Fernandez; Lopez, Jose Vicente; Lyer, Natraj; Machiels, Albert; MacRae, Walter; Malesys, Pierre; Marschman, Steve; Martin, Zita; Massari, John; McCullum, Rodney; McDeavitt, Sean; McMahon, Kevin; McMahon, Mike; Mendez-Torres, Adrian; Metlay, Daniel; Miklos, Marek Dr.; Mitchell, Bob; Modeen, David; Mooney, Mike; Mote, Nigel; Murphy, William; Murray, Paul; Murty, Korukonda Linga (KL); Muthu, Nathan; Nakagome, Yoshihiro; Narayanan, Prakash; Negrini, Teresa ; Nesbit, Steve; Nichol, Marcus; Niyogi, Kalyan; Norwood, Keith; Oberson, Greg; O'Connor, Stephen; Ordaz, Vonna; Ordogh, Miklos; Padovani, Cristiano; Pan, Yi-Ming; Parks, Cecil; Partes, Bettina; Pasamehmetoglu, Kemal; Pechera, Iurii; Pfeifer, Holger; Quecedo, Manuel; Quevedo, David Garrido; Rigby, Doug; Robertshaw, Julian; Robins, Randy; Ross, Peter J., Dr.; Rubenstone, James; Saegusa, Toshiari; Sanjurjo, Manuel Novo; Sataar, Haaroen; Schroeder, Jens; Schwartz, Glenn; Seaman, Craig; Selby, Greg; Sen, Anindya; Seo, Ki-Seog; Serres-Brasch, Aurelie; Sheffield, Richard; Shih, Peter; Shirai, Koji; Sindelar, Robert; Sisley, Steve; Soares, Luiz Antonio Amorim; Soltis, Jeff; Sorenson, Ken; Sowder, Andrew; Sperry, Lee; Stefanovic, Peter; Stipek, Marko; Stockinger, Siegfried; Stockman, Christine; Swift, Peter; Takahashi, Tadayoshi; Tani, Jun-ichi; Tedesco, Giancarlo; Tripputi, Ivo; Twala, Vusi; van der Lee, Jan; Verhoef, Ewoud V.; Versaci, Raul; Vinson, Dennis; Voelzke, Holger; Wagner, John C.; Waldrop, Keith; Wall, James; Wang, Lumin; Wataru, Masami; Waters, Michael; Weiner, Ruth; Wiersma, Bruce; Williams, Jeffrey; Wilson, Ian; Wolff, Dietmar; Wong, Emma; Wood, Peter; Yamamoto, Masahiro; Yamamoto, Tomofumi; Zigh, Ghani; Zuloaga, Pablo; Zurro, Julio Blanco
**Cc:** Campbell, Debbie
**Subject:** DOE's Extended Storage and Disposal R&D gap analysis report now available publicly

*From the desk of John Kessler, manager...*

Dear ESCP members:

For your information, the US Department of Energy's Storage and Disposal R&D Gap Analysis report is now publicly available at:



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http://www.ne.doe.gov/pdfFiles/Gap%20Analysis%20Rev%200%20Final.pdf

John

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**Information Security Reminder**

Information Security Reminder: OpE COMMs contain preliminary information in the interest of timely internal communication of operating experience. OpE COMMs may be pre-decisional and may contain sensitive information.

They are not intended for distribution outside the agency.



**Brett Rini** *(8/26/2008 3:21:04 pm)*

*Revised on 3/21/2012 8:36:23 am*

## Degradation of Palisades Spent Fuel Pool (SFP) Racks

Issue Summary

Palisades has confirmed degradation of their spent fuel pool racks. Initial indications of degradation began in 1988 when the licensee encountered difficulty while inserting a fuel assembly into one of the rack locations. One of the cell walls had swollen and interfered with the fuel insertion. Since 1988, the licensee has identified several other swollen racks because their resident fuel assemblies were unable to be removed. The licensee has not completely verified the cause of the swelling. The licensee believes that the swelling was a result of pressure buildup in the annular space between the inner and outer cell walls because of radiation-induced gas generation (or a possiblyfrom swelling of neutron absorber material).

In 2007, the licensee generated a condition report (CR) to document another stuck fuel assembly. During fuel shuffle in the SFP on July 30, 2007, the fuel handling machine was unable to lift a spent fuel assembly. A hoist overload occurred as soon as the operator attempted to lift the assembly. As of July 30, 2008, the number of cells known to have experienced swelling was 14 total, 11 with fuel stored in them, and 3 empty cells. A summary of each of the stuck assembly incidents is contained in the licensee's apparent cause evaluation.

Spent Fuel Pool Rack Description

The Nuclear Utility Service (NUS) Corporation racks, also known as the Region I racks (because of their location in the pool), were installed in 1977 to increase SFP storage capacity. A rack assembly consists of a rectangular array of storage cells. Each cell consists of two concentric 1/8 inch Type 304 stainless steel cans with Carborundum-manufactured B4C neutron absorber plates installed in the annular gap between the cans. The top and the bottom of the two cans were closed by welding a spacer between the two cans. Originally the design called for the annulus to be water tight. However, cell wall swelling prompted the drilling of 3/16 inch vent holes in the upper region of each cell.

A criticality assessment originally planned for ~ 2011 was moved up following inspector questions about the potential for B4C loss. The licensee attempted to credit SFP boron concentration to compensate; however the Tech Spec design requirement (4.3.1.1) requires that K-effective be less than or equal to 0.95 if fully flooded with unborated water.

A contributing cause for the pressure buildup within the NUS racks is the vent hole size, location, and blockage vulnerability. The SFP racks were purchased without vent holes, but then modified at Palisades. It is suspected that the vent hole of cells with stuck fuel assemblies may be plugged or were mis-drilled during installation. A hole diameter of 0.19" seems to be disputed in several previous corrective action documents. A mislocated hole could result in the hole being drilled into solid material, preventing venting of the cell wall annulus space. In addition, only one small hole was intended to vent all four sides of a cell, a potential problem if cell tolerances limited exchange of gasses between cell walls, or the single hole becomes blocked.

Criticality Assessment

The most notable unknown condition prompted by SFP Region I rack swelling is with respect to the criticality analysis impact that would result from a loss or reconfiguration of B4C material. Although the licensee's criticality analysis of record and TS 4.3.1.1 take no credit for SFP boron in the Region I rack area, in an August 8, 2008 evaluation, the licensee concluded that current (as-loaded) pool configuration K-eff <=0.95 without credit for soluble boron. TS 4.3.1.1, however, assumes the region is filled with new or irradiated fuel assemblies. The most recent licensee assessment determined, based on a number of assumptions, that K-eff would be maintained below 0.95 until rack repairs can be performed. The licensee also determined, for the most reactive previous Region 1 fuel loading configuration, that K-eff remained less than 0.95 if 150 ppm of soluble boron is credited, and K-eff remained less than 1 assuming no soluble boron.

Technical Specification 4.3.1.1.b requires that the SFP racks maintain K-eff less than 0.95 if fully flooded with unborated water. 10 CFR 50.68 permits licensees to credit soluble boron to achieve a K-eff less than 0.95 but requires that the unborated Keff remain less than 1.0 with a probability and confidence of greater than or equal to 95 percent. The licensee concluded that any degree of degradation, up to and including complete loss of neutron absorber, is not expected to result in an increase of K-eff above 0.95 while the SFP boron concentration remains at or above an interim procedural minimum of 2550 ppm (previous minimum, established by TS 3.7.15, was 1720 ppm). The licensee considered the SFP operable but nonconforming. NRC staff did not concur with this assessment. The licensee has since acknowledged that the SFP Region 1 is non-compliant with TS 4.3.1.1 and 10 CFR 50.68. Subsequently, the licensee verbally committed not to permit any SFP activity that could increase Keff and agreed to review and request NRC concurrence to ensure that any contemplated fuel moves will only reduce K-eff, pending restoration of TS and 10 CFR 50.68 compliance. A formal licensee commitment letter is expected.

The licensee's conclusions involve critical assumptions that have not been evaluated by the NRC, but indications are that several assumptions will be challenged.

TS 4.3.1.1 non-compliance continues and is not disputed by the licensee. NRC staff did not conduct an independent criticality analysis but are reasonably comfortable with the licensee assessment of criticality safety as long as no positive reactivity changes are made to Region 1 of the SFP. A docketed licensee commitment letter, expected by August 29, is expected to demonstrate the licensee's understanding of both the regulatory and safety aspects of their SFP degradation and to commit to an aggressive and sufficiently-detailed plan (including milestones and specific actions), acceptable to NRC, to restore full compliance and to maintain SFP criticality safety in the interim. The licensee is expected to restore full compliance in ~7 months in order to support a scheduled March-April 2009 refueling outage which will require positive reactivity changes to Region 1 of the SFP.

The licensee recently performed Boron-10 Areal Density Gage for Evaluating Racks (BADGER) testing on a sample of SFP racks. The licensee was attempting to determine the neutron absorption capability of the chosen racks. A graphic of the current spent fuel pool loading is attached. The cells shown with a faint yellow background have stuck assemblies. The cells that are outlined in blue are the cells that were BADGER tested by the licensee. The licensee was unable to test any of the swollen racks due to either the assemblies being stuck, or the BADGER module's inability to traverse significantly swollen areas.

See the licensee's apparent cause evaluations (2007-stuck assembly and 2008-absorption testing) and operability evaluation report.

**UPDATE**: The licensee submitted a letter of "Commitments to Address Degraded Spent Fuel Pool Rack Neutron Absorber," stating the following:

1. prohibit the movement of fuel assemblies within the SFP that involve positive reactivity changes until fuel storage requirements in TS, 10 CFR 50.68, and the UFSAR are met.

2. prohibit the movement of fuel assemblies within the SFP that involve negative reactivity changes until the NRC has concurred with the planned change.

3. continue to maintain the spent fuel pool between 75°F and 125°F during normal operation in accordance with plant procedures.

4. maintain SFP boron concentration greater than 2550 ppm at all times.

**UPDATE**: The licensee submitted licensee event report (LER) 08-004.

**UPDATE 8/2010:** The licensee submitted a License Amendment Request (LAR) in September, 2009. Because of supplemental information requested by the NRC, the licensee withdrew the LAR in October, 2009. An LAR is planned to be submitted in early 2011 to address the additional information requested. The licensee and the NRC held a pre-submittal meeting in July, 2010 to discuss this LAR. The licensee presentation and meeting summary are available.

Relevant Operating Experience

According to the licensee, these events do not appear to be common to the industry that have NUS racks containing Carborundum. At this point, however, it is unclear to what extent other Carborundum racks have been tested for possible degradation. In the recent Palisades apparent cause evaluation (same as above), the licensee discussed OpE from Connecticut Yankee, Millstone 1, Calvert Cliffs, and Kewaunee. Of those plants, Millstone 1 is the only plant with the same Carborundum plates as Palisades, and the licensee indicated that their test results were satisfactory. It should be noted, however, that, based on the design assumptions at Millstone, any degradation of less than 50 percent would be considered acceptable. Palisades, however, in order to accommodate the desired fuel loading volume and enrichment, assumed 0 percent degradation in their design analyses.

Confirmatory Action Letter issued.

NRC INFORMATION NOTICE 2009-26 **NRC INFORMATION NOTICE 2009-26: DEGRADATION OF NEUTRON-ABSORBING MATERIALS IN THE SPENT FUEL POOL** was issued October 28, 2009.

White Finding for Palisades - see [ML100200720]

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**UPDATE 6/25/2010: TURKEY POINT ALSO HAD A SIMILIAR CONCERN**
The underlying issue here has been addressed by a previous Issue For Resolution (IFR 2008-027 Palisades -Spent Fuel Pool Degradation Criticality Concern - also a white finding) and the resulting Information Notice 2009-26 "DEGRADATION OF NEUTRON-ABSORBING MATERIALS IN THE SPENT FUEL POOL" that has been issued.

**For Turkey Point** - The OFFICE OF ENFORCEMENT NOTIFICATION OF SIGNIFICANT ENFORCEMENT ACTION (EA-10-037) discusses the following: Licensee: Florida Power and Light Company, Turkey Point Nuclear Plant Unit 3, Docket No. 50-250

Subject: ISSUANCE OF FINAL SIGNIFICANCE DETERMINATION AND NOTICE OF VIOLATION AND PROPOSED $70,000 CIVIL PENALTY

This is to inform the Commission of escalated enforcement actions that will be issued on or about June 21, 2010, to Florida Power and Light Company as a result of inspections at Turkey Point Nuclear Plant Unit 3. The escalated enforcement actions consist of a White finding with two associated violations and a Notice of Violation with a proposed civil penalty of $70,000.

The White finding, an issue with low to moderate safety significance which may require additional NRC inspections, involved the licensee□s failure to effectively manage known degradation of Boraflex, a neutron absorber material used in the Turkey Point Unit 3 spent fuel pool. As discussed, there are two violations associated with this White finding:

1. Contrary to 10 CFR 50, Appendix B, Criterion XVI, □Corrective Action,□ the licensee failed to promptly identify and correct a condition adverse to quality. Specifically, in 2004 and 2007, the licensee identified two spent fuel pool storage cells with excessive Boraflex degradation, but failed to take action to correct this condition adverse to quality until identified by inspectors in December 2009.

2. Contrary to Technical Specification 5.5.1.1.a, the licensee failed to maintain the Unit 3 spent fuel storage racks such that the effective neutron multiplication factor (Keff) would remain less than 1.0 when flooded with un-borated water when considering the biases and uncertainties described in the Updated Final Safety Analysis Report. Specifically, dissolution of Boron 10 from Boraflex panels in the spent fuel pool resulted in a reduction in the nominal Boron-10 areal density such that Keff would not have been maintained less than 1.0 if the spent fuel pool had been flooded with un-borated water.

Additional Background information

LER 2502010001R0 - Turkey Point 3: Spent Fuel Storage Design Feature Assumption for Boraflex Degradation is Exceeded.

See ML100700661 for IR 05000250-10-008; 02/14/2010 - 03/05/2010; Turkey Point Unit 3; Problem Identification and Resolution; Preliminary Greater than Green Finding and Potential Escalated Enforcement Violation. 25 Page(s) dated: March 11, 2010.

See ML100840429 04/14/2010 Notice of Public Meeting with Florida Power and Light to Discuss the Safety Significance of One Preliminary Greater than Green finding with two Associated Apparent Violations and the Severity Level of Three AVs Considered for Potential... 6 Page(s), dated: March 25, 2010.

See ML101730313 Letter to FPL on FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING AND NOTICE OF VIOLATION; NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $70,000 (NRC INSPECTION REPORT 05000250/2010009, TURKEY POINT NUCLEAR PLANT)

Note: This issue was also the subject of several news items on 6/23/10:
NRC Proposes $70,000 Fine Against FPL For Turkey Point Violations and FPL Took Issue With NRC Finding.

See hyper-links to the earlier Issue For Resolution (IFR) and the Information Notice for more background information.
Link to IFR 2008-027 Palisades -Spent Fuel Pool Degradation Criticality Concern - (also a white finding).

See **NRC INFORMATION NOTICE 2009-26 DEGRADATION OF NEUTRON-ABSORBING MATERIALS IN THE SPENT FUEL POOL**

This Turkey Point issue therefore screened out of the Issue For Resolution (IFR) process based on earlier agency actions already taken.

Turkey Point issued a supplement to their LER on June 7, 2011.
Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251
Reportable Event: 2010-001-02, Spent Fuel Storage Design Feature Assumptions are Exceeded -Supplement

Closure memo for IFR 2008-027

***This COMM has been posted to the following communities: All Communications, Fuels, Materials/Aging, New Reactors, Shutdown Risk, Spent Fuel Storage & Load Handling***

## Wong, Emma

**From:** Campbell, Debbie [dcampbell@epri.com]
**Sent:** Friday, December 07, 2012 9:42 AM
**To:** Lambert, John; Larson, Ned; Lavara, Arturo; Lavender, Curt; Leblang, Suzanne; Leduc, Daniel; Lesec, Valerie; Levin, Adam; Lin, Bruce; Liu, Yung; Lloret, Miriam; Lombard, Mark; Lombardi, Gianrico; Lopez, F. Javier Fernandez; Lopez, Jose Vicente; Lucas, Matthieu; Machiels, Albert; MacRae, Walter; Malesys, Pierre; Marschman, Steve; Martin, Zita; Massari, John; McCullum, Rodney; McDeavitt, Sean; McMahon, Kevin; McMahon, Mike; Mendez-Torres, Adrian; Merritt, Chuck; Metlay, Daniel; Miklos, Marek Dr.; Mintz, Todd; Mitchell, Bob; Modeen, David; Mooney, Mike; Morris, Kevin; Mostafa, Mostafa Sinnary; Mote, Nigel; Murphy, William; Murray, Paul; Murty, Korukonda Linga (KL); Muthu, Nathan; Nakagome, Yoshihiro; Narayanan, Prakash; Negrini, Teresa ; Nesbit, Steve; Neuburger, Warren; Nichol, Marcus; Niyogi, Kalyan; Norwood, Keith; Nygaard, Adam; Oberson, Greg; O'Connor, Stephen; Ordaz, Vonna; Ordogh, Miklos; Padovani, Cristiano; Pan, Yi-Ming; Parks, Cecil; Partes, Bettina; Pasamehmetoglu, Kemal; Pechera, Iurii; Pfeifer, Holger; Plante, Paul; Quecedo, Manuel; Quevedo, David Garrido; Ridder, Richard; Rigby, Doug; Robertshaw, Julian; Robins, Randy; Ross, Peter J., Dr.; Rubenstone, James; Saegusa, Toshiari; Sanjurjo, Manuel Novo; Sataar, Haaroen; Schroeder, Jens; Schwartz, Glenn; Seaman, Craig; Selby, Greg; Sen, Anindya; Seo, Ki-Seog; Serres-Brasch, Aurelie; Sheffield, Richard; Shih, Peter; Shirai, Koji; Sindelar, Robert; Sisley, Steve; Soares, Luiz Antonio Amorim; Soltis, Jeff; Sorenson, Ken; Sowder, Andrew; Sperry, Lee; Sridharan, Ph.D., Kumar; Stefanovic, Peter; Stipek, Marko; Stockinger, Siegfried; Stockman, Christine; Swift, Peter; Takahashi, Tadayoshi; Tani, Jun-ichi; Tarantino, David; Tedesco, Giancarlo; Teysseyre, Sebastien; Tripputi, Ivo; Tulenko, James "Jim"; Turinsky, Paul J.; Twala, Vusi; Valenta, Heidi; van der Lee, Jan; Vanderniet, Clark; Verhoef, Ewoud V.; Versaci, Raul; Vinson, Dennis; Voelzke, Holger; Wagner, John C.; Waldrop, Keith; Wall, James; Wang, Lumin; Wataru, Masami; Waters, Michael; Weiner, Ruth; Wiersma, Bruce; Williams, Jeffrey; Wilson, Ian; Wolff, Dietmar; Wong, Emma; Wood, Peter; Yamamoto, Masahiro; Yamamoto, Tomofumi; Zigh, Ghani; Zuloaga, Pablo; Zurro, Julio Blanco
**Cc:** Waldrop, Keith; Campbell, Debbie
**Subject:** Preliminary results from NRC/CNWRA SCC experiments

Dear ESCP members:

The NRC presentation on atmospheric salt fog testing given at the NACE conference held in March 2012 is available at NRC's ADAMS:

Website Address: http://adams.nrc.gov/wba/

- Select "Advanced Search"
- Select "Document Properties"
- Click in the space below "Property"
    - From the dropdown box select "Accession Number"
- Click in the space below "Value"
    - Type in: ML120720549
- Select "Search"
- Select the presentation
- Select "Download"

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# Periodic Operating Experience

*POE 2013-02* March 2013

**Inside**



**Operating Experience:**

*The NRC's Operating Experience Center of Expertise reviews events and issues with potentially safety-significant, generic implications for the construction and operation of nuclear reactors.*

## Refueling Outage Season

*Mark King*

Even with the reactor shutdown, outage periods can pose significant risks.

Mid-loop operations in particular at pressurized water reactors pose risk challenges with short time-to-boil and core uncover times.

Other challenges include the high number of infrequently performed evolutions, changes to system configurations and equipment availability, and multiple concurrent activities. All of this happens at a time when defense-in-depth protection present during power operations is reduced.

Inadequate control of reactor coolant system inventory continues to be a problem, despite significant operating experience. This has been an issue at boiling water reactors and pressurized water reactors with several examples just in the past five years (see link).

A particular challenge for maintaining inventory control is ensuring that level indications are accurate. This requires an understanding of the function, requirements, and limitations of the level instrumentation available to the operators.

Careful attention is needed to ensure that multiple level indications are available when possible, especially when making changes to pressurizer or reactor vessel level or venting, and that the status of level instruments is properly recorded and understood.

A list and brief discussion of OpE arising from outage-related activities can be found at this link.

## IOEB Issues Annual TRG Report

*Eric Thomas*

Over 175 NRC staff members contribute to the OpE Technical Review Group (TRG) program. There are 25 TRGs, and each is composed of a team lead (or co-leads) along with group members who are subject matter experts in a given technical area.

When the OpE Clearinghouse screens an issue or event, the team considers which TRGs might be interested in the information, and they promptly forward the issue to the appropriate TRG leads. The TRG leads and members consider this information, along with additional information they collect from other OpE sources or their own division/regional activities, and submit an annual input for their group to IOEB.

In 2012, TRG activities contributed to at least thirteen generic communications. These included IN 2012-13, "Boraflex Degradation Surveillance Programs and Corrective Actions in the Spent Fuel Pool," [Chemistry TRG], Bulletin 2012-01, "Design Vulnerability in Electric Power System," [Electrical Power Systems TRG], and IN 2012-14, "Motor-Operated Valve Inoperable Due to Stem-Disc Separation," [Pump and Valve TRG].

Staff from the Office of New Reactors (NRO) participate in nearly all of the TRGs, and are leads or co-leads for several of the groups. They have a particular interest in issues related to new construction, such as the alkali-silica reaction phenomenon and other structural issues.

A summary newsletter of the 2012 TRG inputs can be found in ADAMS under ML12349A172. Historical TRG results, including the complete inputs from each of the 25 groups, are located on the OpE Sharepoint site.

# Developing Stories

## Pilgrim Extended LOOP Following Winter Storm Nemo

*Dave Garmon*

**Inspector Insights:**

- Phone communications limited to cellular phones (satellite phones remained available)
- Transportation to the site required coordination with state authorities
- Licensee did not implement a manual corrective action program when the electronic version became unavailable during power outage
- EDGs reliably provided vital AC power throughout the event

NRC Task Force Report for 2005 Hurricane Season Lessons Learned

On February 8, Winter Storm Nemo caused a loss of offsite power (LOOP) at Pilgrim. As a result of the LOOP, the licensee declared a Notice of Unusual Event (NOUE). Both of Pilgrim's emergency diesel generators (EDGs) supplied power to vital busses following the LOOP. However, spent fuel pool (SFP) cooling was lost because it is powered from a non-vital bus (pool temperature was 89°F at the onset of the event with about a 90 hour time-to-boil).

On February 9, the licensee restored one of three offsite power lines and repowered the startup transformer (SUT) and certain non-vital buses. After several hours of monitoring the SUT for stability, the licensee shifted vital buses from the EDGs to the restored SUT. On the morning of February 10, the licensee exited the NOUE.

Later that day, the site experienced a second LOOP when a breaker to the SUT tripped. This second LOOP did not result in an NOUE because the plant was already in cold shutdown with two additional sources of power available (2 EDGs and a Station Blackout EDG).

The weather induced breaker fault was attributed to flashover of the 'B' phase post insulator on the SUT. Arcing and sparking above the carrier link of the coupling capacitor voltage transformer (CCVT) on the 'C' phase of the offsite power line resulted in visual damage to certain CCVT components. The licensee applied guidance from a vendor bulletin (ABB Bulletin 2750 514-23, Revision 2) to establish a basis for continued service of the post insulator. In accordance with another vendor manual, the damaged components on the CCVT were removed and an additional ground wire was attached to the CCVT.

As a result of the second LOOP, SFP cooling was lost a second time. The licensee initiated a temporary modification to restore SFP cooling by routing cables from the normal breaker cubicle to a breaker cubicle powered from a vital bus. SFP temperature reached 105°F (a total increase of 16°F) before SFP cooling could be restored a second time.

On February 11, SFP cooling was lost a third time when the diesel driven air compressor failed, resulting in a loss of instrument air which isolated the suction source for the SFP pumps. At this point, plant operators completed aligning the unit auxiliary transformer to provide power to non-vital buses using an available off-site power line. Instrument air and SFP cooling were returned to service once power was restored to all non-vital buses.

## Diablo Canyon Pressurizer Nozzle Indications & Generic Concerns

*Steve Pannier*

Recent In-Service Inspection examinations at Diablo Canyon Unit 2 found that three of the six pressurizer nozzle structural weld overlays (SWOLs) have laminar lack of bond, inter-bead non-fusion flaws. These flaws were revealed using phased array ultrasonic testing (UT), and were found to be initial fabrication issues that were not discovered during initial acceptance testing. The flaw sizes range from approximately 5 inches to completely circumferential in length and 0.25-0.54 inches in width. All of the indications appear on the upstream side of the original base metal welds, in the bevel region of the nozzles (see diagrams here). Previous examinations using conventional UT, including those done as part of an unrelated relief request granted in 2008, had not noted these flaws.

After evaluating the flaws discovered during the recent inspection, the licensee submitted a revision to its alternative relief request, stating that performing required compliance repairs of these flaws would result in hardship or unusual difficulty without increasing the level of quality or safety. NRC staff concurred with the licensee's hardship assessment, stating that there is reasonable assurance that the indications do not challenge the structural integrity of the welds for one cycle of operation. Staff is concerned that the qualification procedures for the conventional and phased array techniques may not be adequate and is evaluating generic aspects of the SWOL UT examination technique qualification.

# Event Follow-Up

## Point Beach Auxiliary Feedwater Issues

*Russ Haskell*

In January 2013, the NRC finalized a white finding for Point Beach 1 citing inadequate work procedures that were used during maintenance activities performed in November 2011 on the turbine-driven auxiliary feedwater (TDAFW) pump. During a subsequent surveillance in May 2012, the pump's flexible coupling failed from excessive stresses caused by misalignment of the exhaust system piping and the turbine flange (LER). In addition, the licensee had machined the turbine bolts in an attempt to alleviate alignment problems experienced while re-assembling the pump (see photo). This activity was not appropriately reviewed and implemented.

Bolt machined in an attempt to align TDAFW pump coupling

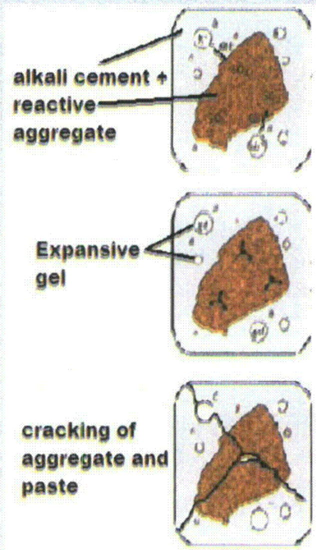
*IN 2008-09 discusses the problems with the TDAFW pump bearing alignment that required Point Beach 1 to shut down.*

In 2002 and 2003, Point Beach received two red findings associated with three violations related to problems with the AFW system. Follow-up inspections found significant deficiencies in the licensee's corrective action program and engineering design control program which resulted in a Confirmatory Action Letter (CAL-3-04-001). A special inspection was chartered in 2007 to evaluate further AFW problems arising from TDAFW pump bearing misalignment (see SIT report).

## Seabrook Alkali Silica Reaction (ASR) Update

*William Raymond (RI)*

NextEra, the licensee for Seabrook, continues engineering evaluations and structural assessments to determine the effects of ASR in concrete structures. These activities are being monitored as part of a larger set of licensee commitments documented in a Confirmatory Action Letter. Current operability of the affected structures is based on a prompt operability determination that was last updated in May 2012. In this update, the licensee asserted that ASR has had a negligible effect on the structural capacity of the affected structures, and that any notable effects can be accommodated by design margin. This determination was sufficient for the licensee to declare the affected structures operable but nonconforming. Testing that is in progress at the University of Texas will be used to update the prompt operability determination. NRC staff concluded that this operability determination was acceptable. Follow-up inspections will review the licensee's testing plan to ensure uncertainties associated with ASR progression are addressed.




ASR Mechanism (adopted from licensee's operability determination ML12151A397 )

NextEra identified two root causes for ASR development: (1) the concrete mix was susceptible; and (2) the monitoring program for plant structures did not contain a process for periodic assessment of failure modes. For example, the structures monitoring program did not consider the possibility of ASR development because the ASR degradation mechanism was assumed not to be a credible failure mode following initial construction. It was later determined that the industry-accepted standards used at the time of construction were poor predictors of slow-reacting aggregates like those used at Seabrook. Although new concrete structures can be constructed of ASR-resistant material, adequate corrective actions for existing structures, as it pertains to concrete mix, are limited to confirming and monitoring the properties of the material in place. NextEra is focusing corrective action efforts on refining its structures monitoring programs be more in line with the industry standard for effective monitoring of Category I structures under 10CFR 50.65, ACI 349.3R. For example, monitoring programs will include periodic re-assessments of failure modes that were initially excluded to ensure initial assumptions are still valid in light of operating experience.

As a result of the ASR issue at Seabrook, licensees for nuclear plants under construction, Vogtle 3 and 4 and Summer 2 and 3 included recommendations proposed in IN 2011-20, "Concrete Degradation by Alkali-Silica Reaction," in their corrective action programs for follow-up. (OpE Comm)

# Developing and Follow-Up Stories Cont'd

## End-of-Cycle RCS Chemistry Impacts RCP Seals

*Bob Bernardo*

*A similar event occurred at Salem in 2006 - LER 3112006003.*

During end of cycle (EOC) coastdown at Vogtle 2 on February 26, the #1 seal leak-off rate for all four operating reactor coolant pumps (RCPs) increased noticeably. The leak-off rate on the #4 RCP exceeded the procedural limits for operation of the RCP, resulting in a manual reactor trip (EN48788).

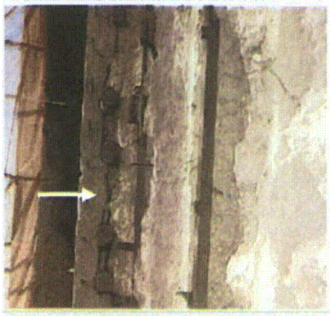
A licensee root cause evaluation concluded that the increased seal leak-off rates were caused by the addition of oxygenated water to the reactor coolant system (RCS) during the change-out of an RCS filter that occurred shortly before the increase in leak-off rates was noted. With little or no boron present in the RCS during EOC operations, the RCS has less ability to buffer the effects of any inadvertent chemistry changes. Increased oxygen concentration caused by the filter change-out triggered a localized crud burst near the RCP seals. The crud burst caused interference at the seal surfaces, resulting in the increased leak-off rates.

All RCS filter change-outs were suspended pending procedure revisions to preclude the addition of oxygenated water to the VCT. Once the pumps were secured, seal leak-off flow returned to normal values with no additional actions needed.

The unique conditions during EOC operations have the potential to aggravate chemistry perturbations that can be introduced during evolutions such as changing Reactor Coolant filters, bypassing or placing in service demineralizers, or operating with deborating resin beds. All of these operations could allow oxygen introduction or cooling of isolated loops, impacting reactor plant chemistry and leading to an unanticipated corrosion product release from associated downstream components.

## Crystal River Containment Delamination Wrap-Up

*Bob Bernardo*



Initial delamination seen in 2009 - note crack running top-to-bottom just left of the tendons

*2009 OpE COMM*

*2011 OpE COMM*

On September 26, 2009, Crystal River shut down for a refueling outage, which was scheduled to include a steam generator replacement. Several horizontal and vertical tendons were detensioned and removed to make an access opening in the containment building. When the tendons were relaxed, the outer layers of the structural concrete surrounding the steel liner of the building delaminated (circumferentially cracked – see picture). After an extensive root cause analysis (see enclosure of SIT report), the licensee removed the cracked portions of the concrete and installed new concrete.

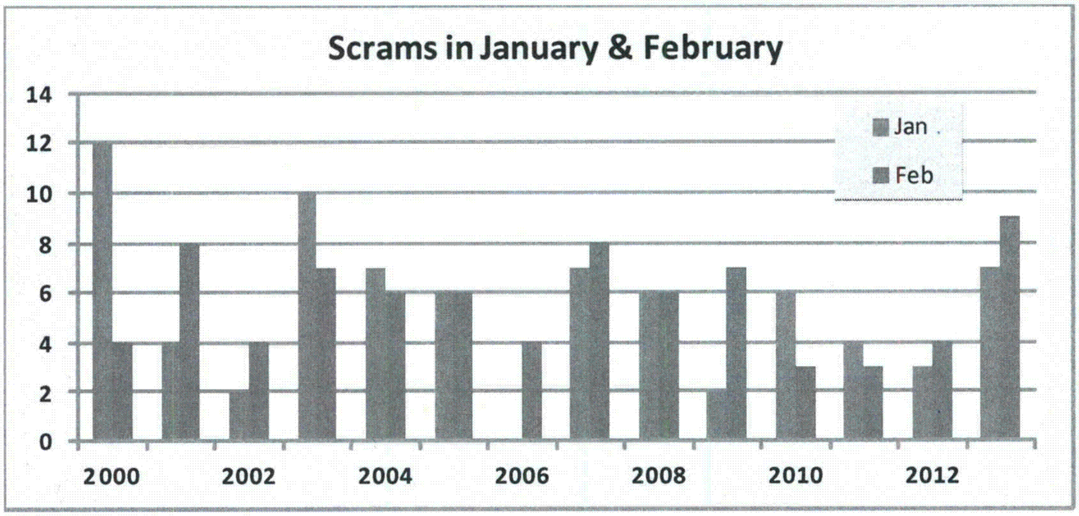
To restore the containment to its full strength condition, the removed tendons were reinstalled and re-tensioned in accordance with a detailed tensioning plan. The plan included the use of acoustic devices to monitor the building during tensioning operations. In March 2011, during the tensioning process, additional significant concrete cracking occurred in other areas of the containment building, beyond where the original cracking had occurred. The licensee halted all repair activities except those required to stabilize the building. From April 2011 until February 2013, the licensee evaluated the scope of the repairs, the risks, and the schedule necessary to successfully repair the containment building. The extensive repairs would require removal of the concrete to the liner over a significant fraction of the containment.

A licensee-initiated engineering report, completed in 2012, determined that repairing the damaged containment structure was viable. However, the nature and potential scope of the repairs had inherent challenges that made cost estimates uncertain. On February 5, 2013, Progress Energy Florida, a subsidiary of Duke Energy, announced that the Crystal River 3 Nuclear Plant would be retired (EN48716).

# 2013 Scrams: Early Analysis

*Joe Giantelli & Rebecca Sigmon*

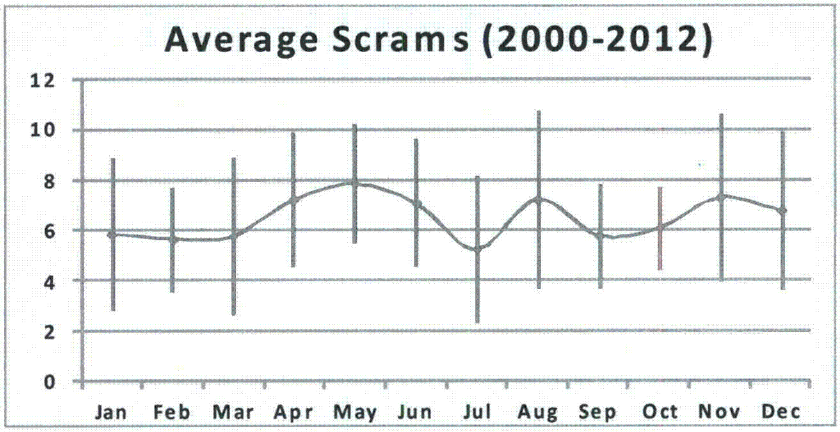
While 2011 and 2012 were record low years for scrams across the industry, the first two months of 2013, with seven and nine scrams respectively, have shown an uptick from recent patterns. February was the first time in over two years where the number of scrams was more than one standard deviation above the 13-year average for that month, but review of each event based on information available shows that no additional Agency action is needed at this time. Analysis of scram patterns since the initiation of the ROP in 2000 shows that scrams are somewhat cyclical in nature. While nine scrams in a month is not particularly unusual (it happened as recently as August 2011), January and February tend to have fewer scrams on average, while the outage months of April-May and November, as might be expected, tend to have more scrams on average. More information is provided in our OpE Note on scrams, found here.




16 scrams in January and February of 2013 came from 12 different plants:

4 plants scrammed twice (South Texas 2, Grand Gulf, Pilgrim, and Turkey Point 3), and 8 plants had a single scram.

*Error bars show one standard deviation from the mean number of scrams in each month. Note that the spread of the standard deviation for each month exceeds the difference between the high and low mean values.*




# First Degraded EP Cornerstone since 2007

*Rebecca Sigmon*

With two white findings in the Emergency Preparedness (EP) cornerstone stemming from the inability to appropriately classify a radiological release (link), Columbia became the first plant to enter Column 3 of the Action Matrix because of a degraded EP cornerstone since DC Cook had a yellow performance indicator in 2007 following a failure of alert and notification system sirens. Columbia also had a degraded EP cornerstone in 2001 following a yellow finding for inadequate protective actions for members of the public (link). Overall, there were five greater-than-green findings in the EP cornerstone in 2012 (all white), the highest number of EP findings with elevated significance in one year since 2002.

DC Cook 95002 Supplemental Inspection Report for Yellow PI in Emergency Preparedness

# OpE Products

## IN 2013-01: Emergency Action Level Thresholds Outside the Range of Radiation Monitors

*Dave Garmon*

IN 2013-01 describes three instances of licensees—Kewaunee, Prairie Island and Crystal River—revising emergency action level (EAL) thresholds to levels outside the monitoring capability of the instruments operators would use to declare the EALs. Each case violated NRC requirements to maintain emergency plans that meet the standards in 10 CFR 50.47 (b)(4). These events highlight the importance of fully evaluating the effects of configuration changes for both equipment and procedures.

*Let us know what you think!*

POE Feedback

## Operating Experience Note: Significant Events and LOOPs

*Rebecca Sigmon*

As part of a larger review of the increase in significant reactor events that has been noticed over the last couple of years, IOEB compiled a high-level summary of what might be driving the increase in OpE Note 002. This includes a more focused review of loss of offsite power events from 2011 and 2012, many of which led to significant events. The two primary insights provided in the summary are:

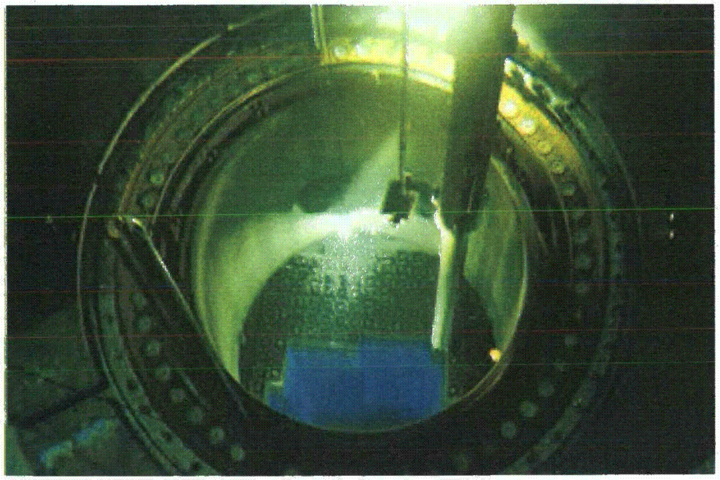
- The increase in significant events in the past 3 years has been driven by an increase in the number of events where an actual initiator is complicated by some combination of equipment failure and inadequate operator response;
- The loss of offsite power events experienced by seven units at three sites in 2011 were caused by natural phenomena beyond the control of the licensees, but none of the five LOOP events in 2012 were caused by natural phenomena.

*See us off the NRR home-page at:*

**OpE Gateway**

*View Previous Issues at:*

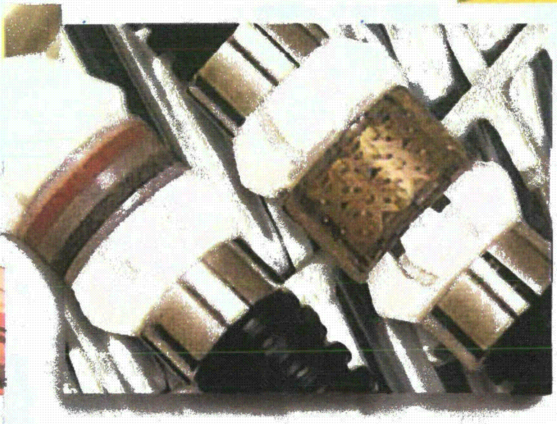
**POE on SharePoint**

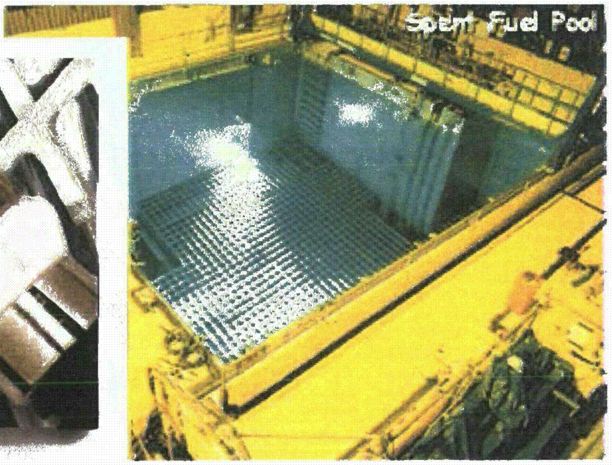


# Guarding the Safety Pillar: Management of Shutdown Risk during Outages

**Outage Risk is <u>Significant</u> Compared to Online Risk:**

- **-7 days of outage ~ 1 year online**
- **-1 day of draining to mid-loop ~ the risk of a yellow finding**
- **-Draining to mid-loop <u>greater</u> risk than being online with loss of both SI pumps, both RHR pumps, all accumulators, 2 offsite lines, and ½ of all remaining safety components**





**Jones, Steve**

| | |
|---|---|
| **From:** | Jones, Steve |
| **Sent:** | Wednesday, March 27, 2013 8:24 AM |
| **To:** | Casto, Greg |
| **Subject:** | RE: RES Presentation on SFPSS |

I don't mean to imply Peach Bottom is unique. I expect most or all BWR owners discharge the spent fuel batch directly to a dispersed configuration now. It may have taken some time to get set up, but now every plant has gone through at least 3 refueling outages with the dispersed requirement in place. It seems to me to be more economical to plan ahead (i.e., open discharge slots during dry cask loading) and get the fuel into the final configuration at refueling than to have to reshuffle.

Steve

-----Original Message-
From: Casto, Greg
Sent: Wednesday, March 27, 2013 8:07 AM
To: Jones, Steve
Subject: RE: RES Presentation on SFPSS

Got it and that agrees more with what I thought. This is related to one of the ongoing comments with the SFPSS, as it uses the 1X4 configuration (or 1X8 in the HRA) immediately including the OCP1, 2, and 3 timeframes for a "reference plant". Peach is unique, and the study doesn't do a good job in exposing that (mentioned, but not connected). I have what I need. Tx greg

-----Original Message-----
From: Jones, Steve
Sent: Wednesday, March 27, 2013 7:54 AM
To: Casto, Greg
Subject: RE: RES Presentation on SFPSS

Greg,

I understand from the SFPSS staff that Exelon does discharge directly into a dispersed configuration, and all BWRs can accomplish that because they normally only discharge the spent batch (about 250-300 assemblies out of 784) to the pool. The BWR owners just need to plan ahead to accomplish that. So the refueling progression in the SFPSS is reasonable. During the infrequent refueling (1 out of 5 or so) involving a full core discharge, immediate dispersal is not likely.

PWRs typically transfer the full core to the spent fuel pool, so the fuel intended to be reloaded is unlikely to less be dispersed and the spent fuel may or may not be dispersed. A number of pools only have about 4 full core's worth of storage space, so those pools cannot physically disperse a full core discharge into a 1X4 pattern.

Steve

-----Original Message-----
From: Casto, Greg
Sent: Wednesday, March 27, 2013 7:33 AM
To: Jones, Steve
Subject: RE: RES Presentation on SFPSS

Thanks, but this is not an accurate description of typical in progress refueling, correct? At best, this is only possible for a partial offload, but not normally arranged in the 1X4 or rarely in the 1X8 until sometime post end of outage. Tx greg



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-----Original Message-----
From: Jones, Steve
Sent: Tuesday, March 26, 2013 4:01 PM
To: Casto, Greg
Subject: RE: RES Presentation on SFPSS

Greg,

I understand the pool configuration slides. The specified days refers to the number of days after reactor shutdown for refueling. Part of the storage configuration reflects the need to load the pool in a manner that MELCOR can accurately model, which is the basis for placing the hottest fuel in the center of the storage array. The 1X4 configuration reflects the storage configuration required by order. At 25 days, refueling is over and the pool has been separated from the refueling canal with the entire batch offload in the center region of the pool. At 4 days, refueling is in progress, the pool has been preconfigured with empty storage cells in a 1X4 configuration, and the first 88 assemblies of the refueling batch have been discharged to the pool. From discussions with RES, the postoutage 1X8 configuration on slide 7 is closer to the actual post outage configuration at the plant.

Steve

-----Original Message-----
From: Casto, Greg
Sent: Tuesday, March 26, 2013 3:32 PM
To: Jones, Steve
Subject: FW: RES Presentation on SFPSS

Quick sanity check. See slide 8, and are you clear on what the 24 and 4 day post outage really means? (post shut down or post re start). Regardless, I would not think that fuel would be in this configuration post outage (even BWR?). Insight? Tx greg

-----Original Message-----
From: Witt, Kevin
Sent: Tuesday, March 26, 2013 3:12 PM
To: Casto, Greg; Reckley, William; Skeen, David
Subject: RES Presentation on SFPSS

Hello all, I found the attached presentation on the RES SharePoint site, which provides a summary of the SFPSS results and is dated March 2013. This may or may not be the presentation that Brian is using to brief the Commissioners.

Thanks,
Kevin

## Anderson, Shaun

| | |
|---|---|
| **From:** | Davis, Jack |
| **Sent:** | Friday, June 07, 2013 4:42 PM |
| **To:** | Anderson, Shaun |
| **Subject:** | FW: Outstanding Issues with SFPSS |
| **Importance:** | High |

FOIA

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**From:** Casto, Greg
**Sent:** Thursday, March 28, 2013 9:25 AM
**To:** Davis, Jack
**Subject:** Outstanding Issues with SFPSS
**Importance:** High

Exec. Summary:

1) #5 – Study assumes that offloaded fuel is immediately placed in 1X4 configuration when removed from reactor. While it is understood that Peach Bottom does that (and also uses a 1X8 configuration), and many BWRs may have the available space to do this nor do PWRs in general do this (and PWRs specifically remain in a contingous pattern for some time post start up. Additionally, the footnote is an example of what used to be security sensitive information.
2) #12 – The amount of land interdiction (single example) used is the highly improbable scenario with a small leak, 40 hours to top of fuel, 57 hours to cladding failure, with unsuccessful mitigation. The consequence example represents a worst case with maximum conditions for hydrogen generation, reactor building containment (not a containment) and oxygen introduction post hydrogen ignition to create the high possible source term. This is not representative of consequences from the identified non-mitigatible scenario where mitigation fails to prevent a zirc fire. (much less offsite consequence). These consequence results use an antiquated MACCS 2 Gaussian dispersion model that is targeted for replacement in future versions. The NRC, industry, and offsite response organization standard for offsite dose assessment is RASCAL, and initial runs using similar source term (to the MACCS run in the SFPSS) only extend PAGs slightly beyond 50 miles. Requests for comparison runs using RASCAL (limited to 50 miles) supplemented by DOE NARAC modeling (unlimited distance) have not been performed to date. It is NRRs opinion that this information is suspect and not validated with standard models. **This is a very important area of the study, and needs more validation before releasing publically.**
3) #13 – The use of permanent interdiction (previously termed comdemnation) is not correctly applied in context with EP principals or EPA guidance. While it is advantageous for health effects studies to evacuate a large extent of the population (and avoid dose to that segment), the outcome for this study describes public relocation from "the reference plant" to Nova Scotia (500 miles downwind). The SFPSS uses a 500 mr/first year public dose criteria for relocation, which is not the EPA 400 guidance dose (2 rem/first year). The SFPSS uses the PA guidance (which is unique in the US to only PA), and discusses the difference in some detail, but the outcome is a much larger footprint.
4) #15 – This is a key mitigation gap identified by the SFPSS. While this was known when developing B.5.b (and not addressed due to low probability) this gap is very evident in this report. Detail within the report on event timing and B.5.b equipment used was evaluated (by NSIR) as not being security sensitive. Staff still question that conclusion.
5) Throughout exec. Summary and section 11 "Conclusions" – There are numerous "sound bites" within the conclusions that state or imply that new insight has been gained by this report (over previous reports). This would conclude that new and different information (NEPA standard) exists, and the WC EIS would be subject to addressing that information. Additionally: fuel is not air coolable in a specific % of the operating cycle (#5) and more favorable loading patterns help to correct (not eliminate ) that (gap in safety? Not required), mitigating



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equip. has largest impact (#7), yet mitigation is not always successful (#8 and #15), specifically for high density SFPs, latent cancer higher (10X) for high density SFPs (#10), land interdiction (land contamination) 100X higher (#12) for high density without successful mitigation, land contamination discussion is new from previous reports (#13), and specific conclusions on what could be stated as deficient areas to past 50.54 (hh) (2) rulemaking.

SFPSS Sections:

1) Section 1.3 – Though the report sites a "reference plant", it is very clear throughout the report that the plant is Peach Bottom. Although some initial information was removed to reduce that tie, much information remains in the report. Consider further removing evidence of Peach Bottom from the report.
2) Section 3.3 – SFPSS uses a .7g earthquake as the example event to fail the pool. NUREG-1353 stated that SFP failure was not possible with less than a 1.4 – 2g earthquake. Though the SFPSS applies only a 10% likelihood, pool failure is still predicted per report. For WC, this may represent one of several "new or different" pieces of information that could challenge their final rule.
3) Section 6.3.4 – Section discusses potential damage to safety related equipment resulting from SFP leak into lower sections of the reactor building. The SFPSS indicates that damage to safety related equipment is possible and identifies the equipment. A recent 2.206 petition response (draft 2.206 director's decision (Lochbaum), reviewed similar events in BWRs and concluded that this outcome is not credible. The SFPSS contradicts that conclusion.
4) Section 7 - Throughout, Level of detail, shows that a lot of consideration went into the analysis and has both good and bad outcomes. As a result of this detail, it builds credibility that the results are correct and validated. I do not believe that they are because 1) MACCS atmospheric model is antiquated (Gaussian model) which by it's design, will greatly expand releases in straight line directions for the duration of the event (48 hours). Newer codes, (RASCAL and IMMAC from DOE) significantly model more realistically and would result in (I believe) far different conclusions. 2) while MACCS is used here to compare health effects with earlier studies (and that makes sense), using MACCS for land contamination outcomes and conclusions (which is new, and not specifically part of earlier studies) is not aligned with NRC, industry, and ORO standards. (RASCAL is standard). Additionally, using the PA guidance for the entire east coast is not appropriate (for a reference plant), and assumptions of use of MACCS for long term interdiction (condemnation) is also not appropriate. 3) Use of the most easily mitigated event (small leak, 40 hr TOF, 57 hr clad damage) to analyze radiological outcomes is not appropriate as a representative consequence. Without a range of analysis at a minimum, including the cases where mitigation gaps exist, the study represents this outcome for all events. This will cause difficulty in explaining "realistic" consequences from possible (and not improbable) non-mitigated events. Overall, this section is not believed to be representative of EP based decision making or valid dose assessment outcomes.
5) Section 8 (HRA) – Section 8.1.1, HRA specifically states that 50.54 make up is insufficient to prevent fire in OCP1. For OCPs 2 and 3, moderate leak scenarios appear to have mitigation failures due to timing to supply mitigating eq. to the pool (all apparent gaps in mitigative protection by B.5.b. This is a significant relelation (though known previously, it was not published for security sensitivity and not addressed due to low probability).
6) Section 8.1.2 – Reasonably low dose rates (in Figure 97) are used as rationale for inability to respond with B.5.b. equipment. NSIR and NRR challenged these assumptions, but they remain. Though I understand that you need to make assumptions, for this public report, this will counter typical ERO response scenario capabilities and boost NGO evidence that response is ill planned and ineffective. RES needs to allow NSIR and NRR to provide valid ERO planned measures to response capability descriptions. This is critical to ability to mitigate several current mitigative gaps.
7) Section 8.3.3.2 – Introduction of delay time is new and not part of B.5.b. criteria. B.5.b. strategies indicate implementation of mitigative strategies within 2 hours of recognition. Delay time, while it may be valid, significantly impacts planned response strategy. This could be seen as another example where NRC actions proved inadequate by NRC. Do we address as gap?
8) Section 9 – Study points out that 1X8 configuration prevents fires, where they occur in 1X4. Again, another ineffective action by the NRC, but according to (other) RES staff, the 1X8 configuration has not been validated in MELCOR (only full drain down single assy and 1X4). Additional user needs are necessary to validate a full range of configurations and drain down scenarios.

9) 9) Page 196, figure 126 – Not previously specifically mentioned, but it appears from data that 200 gpm spray in OCP 2 also does not prevent fire. Overall specific reference for gaps in mitigative strategy need to be identified (this was an old notice that was later dropped, but it shows up in data). It is missing from the HRA, as example.

Overall, this seems to have a far reaching outcome as a public document, and to me, there is something driving release of this that just doesn't appear logical. Data is good, but I do not believe that the staff would stand behind this data and conclusions, as presented. Tx greg

***tx greg***

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# LER Search Results

Return to Previous Search

Displaying **1** to **140** of **140** Records

| Download | LER Number | Report Date | Event Date | Plant | Title/Abstract |
|---|---|---|---|---|---|
| PDF | 4822013005 | 05/13/2013 | 03/13/2013 | Wolf Creek | **Fatigue Failure of Jacket Water Pressure Switch Diaphragm Results in Loss of the 'B' Diesel Generator**<br>*Abstract:* At 0134 Central Daylight Time (CDT) on 03/13/2013, Control Room annunciators 23B, "DG NE02 UV or UF," and 23D, "DG NE02 Trouble," were received for the 'B' diesel generator (DG). At 0149 CDT on 3/13/2013 the Shift Manager declared a Notification of Unusual Event (NUE) for Loss of Electrical Power/Assessment Capability, as both DGs were not available. At the time of the event, the reactor vessel was defueled with all fuel located in the spent fuel pool. The 'A' DG was out of service for maintenance. Power to the safety related busses were being supplied from the offsite power sources.<br><br>The cause of the event was failure of the 'B' DG Jacket Water Pressure Switch (KJPS0162) due to water intrusion in the electrical portion of the switch. Excessive pressure oscillations in the jacket water pressure sensing line led to high cycle fatigue failure of the KJPS0162 diaphragm. The pressure switch was replaced.<br><br>The 'B' DG was returned to service at 0221 CDT on 03/14/2013. The NUE was terminated at 0239 CDT on 03/14/2013. |
| PDF | 5282013001 | 05/07/2013 | 03/08/2013 | Palo Verde 1, Palo Verde 2, Palo Verde 3 | **Unanalyzed Condition due to Spent Fuel Pool Criticality Analysis of Record Not Updated for Power Uprate**<br>*Abstract:* On March 8, 2013, PVNGS engineering personnel determined that certain impacts to the spent fuel pool (SFP) criticality analysis of record (AOR) had not been considered as part of the project to perform a power uprate to 3990 MW thermal in 2003. The power uprate impacted the reactivity of fuel discharged to the SFP but the SFP criticality AOR was not revised to |

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| | | | | account for the increased fuel reactivity. Therefore, this condition is being reported as an unanalyzed condition in accordance with 10 CFR 50.73(a)(2)(ii)(B).<br><br>The cause was procedures and processes lacked adequate rigor to identify impacts to the SFP criticality AOR. Additionally, the impacts of power uprate relative to the SFP criticality AOR were not well known or understood by personnel involved.<br><br>As an interim corrective action, an administrative control was implemented to apply a burnup penalty to low margin fuel assemblies to ensure Technical Specification (TS) reactivity requirements are met under all conditions. Planned corrective actions will revise design change procedures to consider reactivity impacts on the SFP and will revise the SFP criticality AOR using updated methodology and input parameters.<br><br>No similar events have been reported to the NRC in the prior three years. |
| **3232013001** | 04/29/2013 | 02/28/2013 | Diablo Canyon 2 | **Valid EDG 2-1 Start Signal Caused by a Loss of 4kV Class 1E Bus G**<br>*Abstract:* On February 28, 2013, at 21:54 PST, with Diablo Canyon Power Plant (DCPP) Unit 1 in Mode 1 at 100 percent power and Unit 2 shut down and defueled for the Unit 2 Refueling Outage Cycle 17, 4 kilovolt vital Bus G on Unit 2 deenergized, which generated a valid actuation signal to start emergency diesel generator (EDG) 2-1. The EDG did not start, as it was in manual control in preparation for planned maintenance. The planned activity did not anticipate an actual loss of power to Bus G. Therefore, the EDG start signal was valid, and was not part of a pre-planned sequence during testing. DCPP determined this condition was reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A).<br><br>This condition was caused when technicians did not ensure all maintenance prerequisites were met prior to beginning work. DCPP determined the root cause was a lack of a formal process for evaluating the risk of outage emergent work. Additionally, maintenance leadership has not been proactive in their approach to prevent shortfalls in human performance standards and use. DCPP will revise select site procedures to include risk management when shutdown. DCPP will also perform corrective actions to address human performance shortfalls to internalize the use of error prevention tools. This condition did not adversely affect the health and safety of the public. |
| **2852013005** | 04/29/2013 | 02/27/2013 | Fort Calhoun | **Control Room HVAC Modification Not Properly Evaluated**<br>*Abstract:* On February 27, 2013, while reviewing a response to an NRC |

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| | | | | | question, an issue was identified where the modification which moved the control room air conditioners condensers from inside the control room to the auxiliary building roof should have obtained prior NRC approval. The condensers are located in close proximity to one another and are protected by a grating that is not rated to withstand a tornado missile. Therefore, it is possible that both the A and B trains could be struck and rendered inoperable by the same missile. The review determined that prior NRC approval had not been obtained for the modification and the condition was entered in to the station's corrective action program. At the time of discovery, the unit was shutdown with fuel removed.<br><br>A causal analysis is in progress. The results of the analysis will be published in a supplement to this LER. |
| PDF | **4832013002** | 04/15/2013 | 02/13/2013 | Callaway | **Degraded Bearing on 'B' Essential Service Water Pump Motor**<br>*Abstract:* On 2/13/2013, during surveillance testing of the 'B' Train of the ESW system, an Operations Technician noticed that the oil in the sight glass of the lower motor radial bearing appeared darker than normal. Based on analysis of the oil, the 'B' ESW pump was declared inoperable on 2/14/2013 at 0721. Required Action A.1 of TS 3.7.8 was entered. Following replacement of the pump motor due to evidence of a degraded bearing, the 'B' ESW train was restored to operable status at 1345 on 2/16/2013 such that Required Action A.1 was exited after a period of 54 hours and 24 minutes. Based on a conservative evaluation of past operability, it is estimated that the 'B' ESW pump motor would not have been capable of meeting its Operability mission time of 30 days after the August to October 2012 timeframe; therefore, this condition is currently considered reportable. This determination is based on the presence of metallic contaminants found in the oil and on recently increased motor vibration, which are indicative of bearing degradation. The most probable cause is shaft current/electric arcing. Further testing and examination at a vendor facility are required to identify a definitive root cause. Preliminary corrective actions include establishing a preventive maintenance frequency for motor cleaning and refurbishment, which will ensure upper thrust bearing insulation integrity. |
| PDF | **2932013003** | 04/08/2013 | 02/08/2013 | Pilgrim | **Loss of Off-Site Power Events due to Winter Storm Nemo**<br>*Abstract:* On Friday February 8, 2013, at 2117 hours with the reactor initially at 85% core thermal power, Pilgrim Nuclear Power Station (PNPS) experienced a loss of off-site power (LOOP) resulting in a load reject and a reactor scram. All rods fully inserted and the Emergency Diesel Generators |

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| | | | | | automatically started and powered safety-related buses A5 and A6. All other safety systems functioned as required. The plant stabilized in Hot Shutdown. At the time of the event a significant winter storm (Nemo) was buffeting Southern New England. At 2200 hours PNPS in conjunction with the local grid operator determined off-site power sources were not reliable and efforts to restore off-site power were temporarily suspended. At 2200 hours, PNPS declared a Notification of Unusual Event. On February 10, at 1055 hours, one of two off-site power supplies was restored, all safety buses were powered from the startup transformer and the Unusual Event was exited. Later on February 10, at 1402 hours with the plant in Cold Shutdown, ice bridging on a startup transformer insulator caused its 345 KV supply breaker to open resulting in a second LOOP. Again the EDG's started and powered safety-related buses. All other safety systems functioned as required. Shutdown cooling was restored at 1426 hours. On February 10, at 2020 hours, this occurrence was reported to the USNRC as documented in EN# 48739.<br><br>The severe winter storm which caused extensive generalized geographical damage to the electrical distribution network was root cause of the LOOP events.<br><br>These events posed no threat to public health and safety. |
| | **2852012020** | 01/31/2013 | 12/02/2012 | Fort Calhoun | **Raw Water Pump Anchors**<br>*Abstract:* On December 2, 2012, while in Mode 5 (De-fueled), Fort Calhoun Station (FCS) determined that raw water pumps (AC-10A/B/C/D) base plate support anchors were not to be in accordance with design requirements due to of inadequate embedment. This resulted in the inoperability of all four pumps and a violation of Technical Specification requirements during past operating cycles. On January 9, 2013, FCS completed calculation FC08216, Rev 0, Raw Water Pump AC-10A/B/C/D Ultimate Failure. This calculation, without safety/reductions factors, resulted in lower tensile loading requirements during a seismic event and no failure of the anchors. To return the base plate support anchors to design requirements, raw water pumps AC-10A/B/C base plate support anchors have been replaced with maxi bolts. Pump AC-10D repairs are pending. The cause has been determined to be FCS Engineering personnel failing to validate the actual plant configuration and the use of uncorroborated drawing information in completion of design basis calculations. |
| | **2932012003** | 12/31/2012 | 10/31/2012 | Pilgrim | **Both Trains of Standby Gas Treatment System Inoperable**<br>*Abstract:* On Wednesday, October 31, 2012 at 1200 hours, with the reactor mode switch in RUN at approximately |

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| | | | | | 100 percent core thermal power and steady state conditions, Standby Gas Treatment (SBGT) System Train "B" was removed from service (made inoperable) for surveillance testing. At 1441 hours, the control room staff declared the SBGT System Train "A" inoperable as a result of an engineering analysis that determined that 480 VAC feeder breaker to Motor Control Center (MCC) B15 had the potential to exceed its trip set point under the worst case bus loading. The inoperability of both SBGT System Trains "A" and "B" could have prevented the fulfillment of the safety functions to "control the release of radioactive material" and "mitigate the consequences of an accident". At 1510 hours, a compensatory measure was taken to preclude the potential overload condition on MCC B15 and the SBGT System Train "A" was restored to operable status to fulfill the safety functions to "control the release of radioactive material" and "mitigate the consequences of an accident". This event had no impact on the health and/or safety of the public. |
| | **3902012005** | 12/15/2012 | 10/16/2012 | Watts Bar 1 | **Automatic Start of Emergency Diesel Generators due to Failed Transfer of Power to 6.9kV Shutdown Board** *Abstract:* On October 16, 2012, at 2330 EDT, Watts Bar Nuclear Plant (WBN-1) licensed operators attempted a manual fast transfer of the 1B-B 6.9kV Shutdown Board (SDBD) from the normal feeder breaker to the alternate feeder breaker. The transfer was not successful, resulting in the automatic start of the four Emergency Diesel Generators (EDGs). After the 1B-B 6.9kV SDBD de-energized and the loads were shed, the alternate feeder breaker closed and re-energized the 1B-B 6.9kV SDBD. The loads supplied by the 1B-B 6.9kV SDBD were subsequently reconnected, and required tests were successfully completed to ensure operability of the 1B-B 6.9kV SDBD. At the time of the event, WBN-1 was in MODE 5 following a refueling outage. Operations personnel promptly entered the appropriate response procedure and re-established power to required loads. Required safety systems functioned as designed. This condition did not adversely affect the safe operation of the plant or the health and safety of the public. The cause of this event was that plant operators did not ensure the alternate feeder breaker hand-switch was held firmly in the "closed" position while initiating the fast board transfer. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), a condition that resulted in automatic actuation of the EDGs. |
| | **3332012005** | 12/04/2012 | 10/05/2012 | FitzPatrick | **Transformer Installation Error Causes Loss of Off-Site Power** *Abstract:* On 10/5/12 at 1301, the James A. FitzPatrick Nuclear Power Plant experienced a loss of off-site power. This event occurred after both Reserve |

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| | | | | Station Service Transformers, 71T-2 and 71T-3, were replaced during Refueling Outage 20. Several hours after installation a maintenance activity which applied a load to the transformer caused a trip of 71T-3 resulting in the loss of off-site power. Investigation identified that the phase A differential protection relay, 71-87-A-1RSSA01, for 71T-3 tripped because the shorting bars (a factory setting) were not removed during installation. The loss of off-site power resulted in a loss of Reactor Protection System power which caused an automatic Primary Containment Isolation System isolation of Reactor Water Clean-Up and Drywell floor and equipment drains. The Emergency Diesel Generators started but one EDG output breaker did not close. In addition, the loss of power caused a loss of Emergency Response communications response capability. This event was reported to the NRC by ENS 48386. The root cause was determined to be not following the work order instructions as written. A contributing cause was an incorrect design drawing. |
| **5282012004** | 10/29/2012 | 08/29/2012 | Palo Verde 1 | **Essential Spray Pond Pump Actuation Due to a Control Room Essential Filtration Actuation Signal** *Abstract:* On August 29, 2012, the Unit 1 control room received a fuel building ventilation exhaust radiation monitor 1JSOBRU0145 (RU-145) high radioactivity alarm. This resulted in actuation of the train A and B fuel building essential ventilation actuation signals (FBEVAS) and control room essential filtration actuation signals (CREFAS). The CREFAS started the train A and B control room essential air filtration units, essential chilled water systems, essential cooling water systems and essential spray pond systems. Alternate sampling and radiation monitor comparisons determined the RU-145 high radioactivity alarm to be invalid. An investigation determined the RU-145 high radioactivity alarm was caused by failure of a power supply zener diode and resultant loss of the 24 VDC low voltage power supply. Loss of the 24 VDC supply activated the check source feature which raised the radiation monitor output to above the high alarm set-point value.The faulty power supply was replaced. No additional actions were determined to be necessary because existing preventive maintenance requirements replace the power supply board every 7.5 years and zener diodes are reliable in voltage regulation applications for the radiation monitoring system at PVNGS. This was the first failure of this type at PVNGS with greater than 25 years of operation. In the past three years, PVNGS has not reported a similar event to the NRC. |
| **2892012002** Revision 0 | 02/22/2013 10/09/2012 | 08/10/2012 | Three Mile Island 1 | **Missing Seals in Air Intake Tunnel Conduits** *Abstract:* On 08/10/12 a TMI-1 flood inspection walkdown discovered that conduits carrying cabling from yard |

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| | | | | electrical vaults through the Air Intake Tunnel (AIT) to the Auxiliary Building (AB) did not contain internal seals for flood protection. When the plant was constructed in the early 1970's the conduit seals were never installed. The conduit seals are internal to the conduit fittings and not visible externally. The cause has been attributed to inadequate configuration management during original construction. The corrective action to modify the design to provide the required flood protection for AIT conduits was completed in October 2012. The safety significance of the past condition was evaluated. Had a probable maximum flood (PMF) event occurred at TMI-1, after a greater flood hazard was recognized in September 2011 and before the AIT conduit deficiency was identified and temporarily mitigated in August 2012, it is likely that some safe shutdown equipment would have been adversely affected but safe shutdown conditions would have been maintained and there would not have been any adverse impact on public health and safety. The submittal of this LER constitutes reporting to the NRC in accordance with 10 CFR 50.73(a)(2)(v)(B). |
| **3952012002** | 08/07/2012 | 06/14/2012 | Summer | **Seismically Qualified Refueling Water Storage Tank Aligned to Non-Seismic Piping**<br>*Abstract:* On 06/14/2012, with the plant in Mode 1 at 100% power, it was determined that opening the code boundary valve between the safety related and seismically qualified Refueling Water Storage Tank (RWST) and the non-safety related and non-seismically qualified Spent Fuel Pool (SFP) Purification Loop in Modes 1-4 renders the RWST inoperable. This alignment was utilized for RWST water mixing in support of to weekly surveillance sampling and for filtration of the RWST water prior to refueling outages. As a result, on multiple occasions the RWST was inoperable for a period longer than allowed by Technical Specifications (TS) 3.5.4, Emergency Core Cooling Systems - Refueling Water Storage Tank, Limiting Conditions for Operation (LCO). The cause of this event is a result of regulatory requirements for the separation of seismically qualified and non-qualified systems, structures and components not being adequately incorporated into the Design Basis Document (DBD) and Updated Final Safety Analysis Report (UFSAR). Immediate actions consisted of implementation of a Station Order (11-22), which indefinitely suspended this alignment, and submittal of a license amendment request (LAR) to revise TS 3.5.4 such that the non-seismically qualified piping of the SFP Purification System may be aligned to the RWST by operation of a seismically qualified manual ASME code boundary valve under administrative controls for performance of RWST surveillance |

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| | | | | requirements and pre-outage filtration. This change will only be applicable through the next two fuel cycles. |
| 3462012001 | 07/23/2012 | 05/19/2012 | Davis-Besse | **Direct Current Source for Diesel Generator Transferred to Inoperable Source during Fuel Movement** *Abstract:* On May 19, 2012, with the Davis-Besse Nuclear Power Station in Mode 6 and movement of irradiated fuel in progress, the Direct Current (DC) power source providing the loss of power start function for the one required Emergency Diesel Generator (EDG) was transferred from its alternate to normal source. While the normal source was functional and available, required surveillance testing had not shown the DC power source to be operable following replacement of a battery cell and completion of a performance discharge test. This deficiency was identified during preparations for reloading fuel into the reactor core on May 22, 2012. The cause of this event was determined to be less than adequate administrative controls for maintaining the DC System power source operability with the system cross-tied during shutdown conditions. Subsequent testing showed the equipment was operable, and procedures will be revised to add a prerequisite for ensuring operability of the motor control center being transferred to, or to ensure both EDGs are operable. This event is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B) as operation of the plant in a condition prohibited by the Technical Specifications and 10 CFR 50.73(a)(2)(v)(D) as a condition that could have prevented fulfillment of a safety function for a system needed to mitigate the consequences of an accident. |
| 3542012004 Revision 0 | 12/10/2012 07/03/2012 | 05/10/2012 | Hope Creek | **As-found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable** *Abstract:* On May 10, 2012, PSEG received the initial results for the safety relief valve (SRV) pilot valve 'as-found' setpoint testing. The results indicated that two SRV pilot valve setpoints exceeded Technical Specification (TS) allowable tolerance specified in TS 3.4.2.1. This specification requires SRV setpoint limits to be within +/- 3% of the specified value. The valves failing to meet limits were Target Rock Model 7567F two-stage SRVs. As planned all 14 SRV pilot valves were removed and replaced with pre-tested, certified spare pilot valves during refueling outage H1R17. All 14 SRV pilot valves were 'as-found' tested at an offsite test facility. A total of six of the 14 SRV pilot valves experienced setpoint drift outside of the TS 3.4.2.1 limits. Five of the six SRVs were within the maximum allowable percent increase (MAPI) value. The SRV-F was the only SRV that did not meet the MAPI value. A Technical Evaluation assessed whether the stresses imposed by the increased lift setpoint would have |

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| | | | | been below the ASME Section III, Appendix F value for failure. The results of the Technical Evaluation are being communicated in this supplemental LER. The cause of the setpoint drift for all six SRVs is corrosion bonding, which is consistent with industry experience. The materials combination for the pilot disc and the pilot seat has been a known industry issue because of the design of the Target Rock 2 stage SRV. This condition is reportable under 10CFR50.73(a)(2)(i)(B) as any operation or condition prohibited by the plant Technical Specifications. |
| **3822012004** | 06/21/2012 | 04/22/2012 | Waterford 3 | **Essential Chiller Oil Leak Creates Unanalyzed Past Operability Condition**<br>*Abstract:* At 14:43 CDT on 04/22/2012, a 22 drop per minute oil leak was discovered on instrument tubing for Essential Chiller A Compressor Low Oil Pressure Switch. Due to the noted oil leakage and loss of compressor oil inventory, the Essential Chiller Technical Specification (TS) 72 hour shutdown action was entered. Plant personnel aligned Essential Chiller AB (swing chiller) to replace A. Essential Chiller Train A was declared operable at 00:44 on 4/23/2012. Examination of the leak revealed that the tubing had been tie-wrapped to adjacent piping. Friction from normal system vibration had worn through the tubing. The instrument tubing was subsequently replaced. In evaluating the past operability of the Essential Chillers within the 30 day mission time, several occasions were discovered where opposite train equipment had been declared inoperable. This created a condition where both trains of Essential Chillers were inoperable, which is not covered by actions in the Essential Chiller TS. As none of the individual periods where both trains were inoperable exceeded 24 hours, consistent with the mission time assumed in quantitative risk analysis, the safety significance of the condition is negligible. This condition is being reported pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B), 10 CFR 50.73(a)(2)(ii)(B), and 10 CFR 50.73(a)(2)(v)(D). |
| **2612012002** | 05/09/2012 | 03/16/2012 | Robinson 2 | **Unplanned LCO 3.5.4 Entry Due to RWST Alignment to Purification**<br>*Abstract:* The Refueling Water Storage Tank (RWST) was placed on purification in accordance with OP-913, Refueling Water Purification Pump Operation, as directed from OP-301-1, Chemical And Volume Control System (Infrequent Operation), at 04:00 on March 16, 2012, to support make up of level to the RWST. The Plant was in MODE 4 with the Reactor Coolant System at approximately 285 degrees Fahrenheit. This condition, connection of the purification loop, is not currently allowed based on unresolved seismic concerns with purification piping to the RWST. This was later discovered during a log review at 05:45, and operators were |

immediately directed to remove the RWST from purification. Technical Specification (TS) 3.5.4 was applied from 04:00 based on when it was determined that this condition had been entered. TS 3.5.4 was exited at 06:22 when the RWST was removed from purification. The cause of this event was determined to be a result of ineffective implementation of previous corrective actions from HBRSEP, Unit No. 2 Condition Report (CR) 463557 and reported in LER 2011-001-00. Operating procedures associated with placing the non-seismic Refueling Water Purification (RWP) loop in service on seismic systems were suspended and Caution Tags (CTs) on SFPC-805A, RWP Pump Suction Isolation Valve from RWST, and SFPC-805B, RWST Return Isolation Valve were replaced with DangerTags which state "DO NOT OPERATE." The condition described in this Licensee Event Report is reportable in accordance with 10 CFR50.73(a)(2)(i)(B), Any operation or condition which was prohibited by the plant's Technical Specifications and 10 CFR 50.73(a)(2)(v)(D), Event or Condition that could have prevented fulfillment of a safety function.

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| **2472012002** Revision 0 | 11/19/2012 04/13/2012 | 02/17/2012 | Indian Point 2 | **Technical Specification (TS) Prohibited Condition Caused by New Fuel Assemblies Stored in a Configuration Prohibited by the TS** *Abstract:* On February 17, 2012, a Fuel Transfer Form preparer using Technical Specification (TS) 3.7.13 (Spent Fuel Pit Storage) as a reference, recognized an error in the Transfer Form (2-TF-2012-004) that had allowed 11 fresh fuel assemblies (FA) to be moved into the Spent Fuel Pool (SFP) on January 23 thru 24, 2012, in Region 1-1 locations in a face-adjacent location to spent FAs. This configuration was not permitted by TS 3.7.13, which requires empty cells in between the Region 1-1 checkerboard locations of fresh fuel. The direct cause was that Reactor Engineering (RE) move sheets issued in January 2012 were incorrect. The root cause was weak/ineffective use of Human Performance (HP) tools during preparation and verification of the move sheets. Both the preparer and verifier did not review TS 3.7.13 in its entirety. The error was a result of poor self and peer check/review, overconfidence during performance of the task and weak supervisory oversight due to failure to perform a pre-job brief. Corrective actions included preparation of a new Transfer Form, disciplinary action which suspended qualifications of the preparer and verifier, performance of a Level 1 HP error review and stand-down to reinforce expectations for TS compliance, and performance of a HP Engineering Department clock reset. A training plan was developed to reinforce expectations of TS compliance and independent verification and self checking. The event had no significant effect on public health and safety. |

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| | **3482012001** | 04/09/2012 | 02/15/2012 | Farley 1, Farley 2 | **Seismically Qualified RWST Aligned to Non-Seismic Piping**<br>*Abstract:* On February 15, 2012, with both Units 1 and 2 operating 100 percent power, it was determined that opening the boundary valve between the safety related and seismically qualified Refueling Water Storage Tank (RWST) and the non-safety related and non-seismically qualified Spent Fuel Pool Purification (SFPP) system in Modes 1-4, renders the RWST inoperable. Plant procedures had been revised in 2009 to allow opening this boundary valve in Modes 1-4 under administrative controls. The 10 CFR 50.59 safety evaluation that had been performed to support the procedure change had concluded that the administrative controls would allow the RWST to remain operable. However, in consideration of the new interpretation provided in NRC Information Notice 2012-01, it was judged that the RWST would be considered to be inoperable regardless of the administrative controls established when the RWST was aligned to non-seismic piping in Modes 1 - 4. Since the boundary valve had been opened in Mode 1 under administrative controls and the one hour completion time of Technical Specification 3.5.4 Condition B was not entered, under this recent interpretation, this represented a condition prohibited by Technical Specifications and is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B). This event had no significant safety consequence since a seismic event had not occurred while the SFPP system was in service on the RWST. |
| | **4242012001** | 04/11/2012 | 02/15/2012 | Vogtle 1 | **Seismically Qualified RWST Aligned to Non-Seismic Piping**<br>*Abstract:* On February 15, 2012, with the unit at 100 percent power, it was determined that opening the boundary valve between the safety related and seismically qualified Refueling Water Storage Tank (RWST) and the non safety related and non seismically qualified Spent Fuel Pool Purification (SFPP) system in Modes 1-4, renders the RWST inoperable. Plant procedures had been revised in 2009 to allow opening this boundary valve in Modes 1-4 under administrative controls. The 10 CFR 50.59 safety evaluation that had been performed to support the procedure change had concluded that the administrative controls would allow the RWST to remain operable. However, in consideration of the new interpretation provided in NRC Information Notice 2012-01, it was judged that the RWST would be considered to be inoperable regardless of the administrative controls established when the RWST was aligned to non-seismic piping in Modes 1-4. Since the boundary valve had been opened in Mode 1 under administrative controls and the one hour completion time of Technical Specification 3.5.4 Condition D was not entered, under this recent interpretation, this represented a |

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| | | | | condition prohibited by Technical Specifications and is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B). This event had no significant safety consequence since a seismic event had not occurred while the SFPP system was in service on the RWST. |
| **2862012002** | 04/13/2012 | 02/13/2012 | Indian Point 3 | **Technical Specification Prohibited Condition due to Exceeding the Allowed Completion Time for an Inoperable Refueling Water Storage Tank during Connection to Purification System**<br>*Abstract:* On February 13, 2012, a review of NRC Information Notice (IN) 2012-01 (Seismic Considerations-Principally Issues Involving Tanks) determined there was a clarification in NRC's position regarding aligning non-seismic piping to the seismically qualified Refueling Water Storage Tank (RWST). The IN identified failures by licensees to recognize that aligning non-seismic piping to the RWST would require Technical Specification (TS) Limiting Condition of Operation (LCO) action statement entry or license amendments. Intentionally aligning the seismically qualified RWST piping to non-seismic Fuel Pool Purification System (FPPS) by opening a boundary valve can cause the RWST to become inoperable. TS LCO 3.0.2 requires that upon discovery of a failure to meet an LCO, the required actions of the associated conditions must be met. TS LCO 3.0.2 does not allow applying compensatory measures such as manual actions in place of a closed boundary valve for periods longer than the TS completion time for restoring the RWST to operable unless the TS expressly permit such operation. Indian Point unit 3 had performed a safety evaluation based on IN 97-78 that allowed operator action. IN 2012-01 clarifies that application of compensatory actions for periods longer than the TS completion time is not allowed. The apparent cause was historical interpretation. Original issue was resolved using the NRC guidance for operator manual actions. Corrective actions will include revision of system operating procedure 3-SOP-SI-003 to prevent aligning the RWST to the FPPS during applicable Modes until the issue is resolved, evaluate the feasibility of a license amendment to allow operator action, assessment of FPPS piping for upgrade to seismic. The event had no effect on public health and safety. |
| **4832011007** | 01/10/2012 | 11/13/2011 | Callaway | **Inadvertent Non-Compliance With TS 3.9.2, Unborated Water Source Isolation Valves**<br>*Abstract:* Per Technical Specification (TS) 3.9.2, all unborated water source isolation valves that are connected to the Reactor Coolant System must be secured in the closed position to prevent unplanned boron dilution of the reactor coolant during Mode 6. While the plant was in Mode 6 on 11/13/2011, one of the valves used to isolate an unborated |

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| | | | | | water source was found to be closed but was not secured in that position. The valve was in this condition when the plant entered Mode 6 from "No Mode," but this was not identified until after subsequent completion of loading fuel into the reactor vessel. The plant entered Mode 6 on 11/07/2011. No core alterations were in progress when the valve was found to be unsecured. Having the valve closed but not secured during Mode 6 (and without meeting the Required Actions of Condition A under LCO 3.9.2) is a condition prohibited by the TS. After the valve was found to be unsecured, the valve was secured in the closed position. Additionally the reactor coolant system boron concentration was verified to be within TS limits. Procedure OSP-BL-00001 did not include adequate instructions to control the status of valve BGV0601 in "No Mode." Plant procedures will be revised to clarify the administrative guidance. Additionally, operations personnel were coached on requirements for procedure compliance. |
| | **2632011008** Revision 0 | 02/28/2012 12/19/2011 | 10/21/2011 | Monticello | **Reactor Scram due to Loss of Normal Offsite Power** *Abstract:* At 1250 on October 21, 2011, at the Monticello Nuclear Generating Plant, a 2R Auxiliary Transformer lockout unexpectedly occurred causing off-site power to automatically transfer to the 1R Auxiliary Transformer, which resulted in a reactor scram. One cable of the "A" phase conductor, supplying power from 2RS to 2R Transformer, faulted to ground, resulting in the 3N4 breaker opening, as designed, to protect 2RS Transformer and other equipment from fault current damage. Subsequent testing indicates the cable suffered from environmental and age-related degradation. Implementation efforts to replace the cables between 2R Transformer and 2RS Transformer were in progress at the time of the event. A portion of the new raceway was under construction. The cables were replaced entirely employing a route designed to avoid cable submergence in water. Subsequent to installation, cables were successfully tested and returned to service. |
| | **2852012008** Revision 0 | 11/29/2012 07/27/2012 | 09/28/2011 | Fort Calhoun | **Technical Specification Violation for Fuel Movement (VA-66)** *Abstract:* On September 28, 2011, Fort Calhoun Station (FCS) Condition Report 2011-7800 identified the failure of the spent fuel pool area charcoal filter (VA-66) to pass the elemental iodine removal test. During a subsequent review of this CR by the Recovery Engineering group, it was determined that on June 6, 2012, fuel had been moved during a time when VA-66 was required to be OPERABLE. The FCS Technical Specification, 2.8.3(4), requires the Spent Fuel Pool Area ventilation system to be IN OPERATION during REFUELING OPERATIONS. A cause analysis determined that a lack of management oversight and the failure of Engineering to take a proactive |

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| | | | | | approach in the prevention of future test failures led to this event. Completed corrective actions include 1) a revision of the applicable procedure to trend charcoal sample results and predict replacement 2) replacement of the depleted charcoal currently installed, and 3) a change the frequency of the charcoal testing from eighteen months to 1 year. |
| | **4822011010** | 10/31/2011 | 09/01/2011 | Wolf Creek | **Diesel Generator Declared Inoperable Due to Inadequate Adjustment of the Diesel Generator Governor** *Abstract:* On September 1, 2011, during the performance of procedure STS KJ-005A, "Manual/Auto Start, Sync and Loading of EDG NE01," the kilowatt (kW) output, current output and the fuel racks oscillated excessively at full load on the "A" Diesel Generator (DG). The peak-to-peak oscillations in kW output observed in the Control Room were 300 to 400 kW. The "A" DG was declared inoperable at 1443 Central Daylight Time (CDT) on September 1, 2011. The cause of the oscillations was due to an adjustment made to the "A" DG governor in May 2011. The governor was restored to its previous adjustment and the "A" DG was returned to service at 1032 CDT on September 4, 2011. The "B" DG and both offsite circuits were operable while the "A" DG was inoperable from 1443 CDT on September 1, 2011 through 1032 CDT on September 4, 2011. Further evaluation of this event is in progress and includes determining whether the "A" DG was capable of performing its specified safety function from May 21, 2011 through September 4, 2011. |
| | **2952011001** | 09/06/2011 | 07/14/2011 | Zion 1, Zion 2 | **Improper Storage of Fuel Rod Storage Canister in Spent Fuel Pool** *Abstract:* On July 13, 2011, during a fuel data management software program site training class, it was identified that a Fuel Rod Storage Canister (FRSC) was stored in the wrong region of the Spent Fuel Pool (SFP). A fuel reconstitution campaign was conducted in 1992. Thirteen damaged fuel rods from seven different fuel assemblies were placed in the FRSC designed by Westinghouse and stored in the SFP during that campaign. At that time the SFP contained a single region rack design with no fuel assembly storage restrictions based on initial enrichment and fuel burnup. In 1993, a new SFP rack design utilizing a two region configuration was installed. The design included Region 2 storage restrictions based on the initial fuel enrichment and fuel burnup in accordance with Permanently Defueled Technical Specification (PDTS) 3.1.3, "Spent Fuel Assembly Storage." Following installation of the new rack design, the FRSC was placed in a SFP Region 2 rack cell. The criticality analysis for the FRSC contains restrictions on the storage location based on the most limiting rod stored within the FRSC. This means that if the |

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| | | | | | FRSC contains rods from a fuel assembly with SFP region storage restrictions, then those same restrictions apply to the FRSC. A review of records conducted on July 14, 2011 determined that nine of the thirteen fuel rods stored in the FRSC did not satisfy the requirements to allow Region 2 rack storage. Consequently, storage of the FRSC in SFP Region 2 since 1993 is considered a violation of Permanently Defueled Technical Specification (PDTS) 3.1.3. Upon discovery, immediate actions to relocate the FRSC to SFP Region1 were initiated. The FRSC relocation was completed on July 20, 2011. A subsequent technical evaluation has concluded that Region 2 storage of the FRSC was within the rack design basis criticality limits. |
| | **2772011002** | 07/29/2011 | 06/06/2011 | Peach Bottom 2, Peach Bottom 3 | **Condition Prohibited by Technical Specifications due to Degraded Spent Fuel Pool Racks Boraflex® Panels**<br>*Abstract:* On 6/6/11, the plant staff determined that certain fuel assemblies in the Unit 2 Spent Fuel Pool (SFP) racks needed to be relocated to other Unit 2 SFP rack locations to gain additional fuel assembly subcriticality margin. This determination was based on information provided by the Nuclear Regulatory Commission (NRC) as part of the development of an NRC Task Interface Agreement (TIA). The TIA was performed as a result of an NRC unresolved item involving the plant staff's assessment of the degradation rate of the SFP rack neutron absorbers (i.e., Boraflex®). This event is considered to be a condition prohibited by Technical Specifications (TS) since additional SFP subcriticality margin was determined to be appropriate to meet TS design requirements for the SFP racks. The cause of the event was due to a degradation of the SFP rack neutron absorbing material (i.e., Boraflex®). The previous station assessment for the rate of degradation of Boraflex® was found by the NRC, as part of their TIA, to not have been conservative enough. Modifications to both the Unit 2 and Unit 3 SFPs will be performed. No subcriticality margin concerns existed on the Unit 3 SFP. There were no actual safety consequences associated with this event. Reduced margin in SFP racks containing Boraflex® is an industry concern and has been the subject of previous NRC generic correspondence. |
| | **2612011001** | 07/05/2011 | 05/04/2011 | Robinson 2 | **Condition Prohibited by Technical Specifications When Non-Seismic System Was Aligned to Refueling Water Storage Tank due to Regulatory Requirements Not Adequately Incorporated in Plant Documentation**<br>*Abstract:* On May 4, 2011, with H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, in Mode 1 at 100% power, it was determined that over the last 40 years, HBRSEP, Unit No. 2, periodically performed cleanup of the Refueling Water Storage Tank (RWST) by aligning |

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| | | | | | the non-seismically qualified refueling water purification system to the safety related and seismically qualified RWST without recognizing that the action rendered the RWST inoperable. As a result, on multiple occasions, the RWST was inoperable for a period longer than allowed by Technical Specification (TS) Limiting Condition for Operation 3.5.4, EmergencyCore Cooling Systems - Refueling Water Storage Tank. The cause of this event was that regulatory requirements for the separation of seismically qualified and non-qualified systems, structures, and components were not adequately incorporated into the Design Basis Document (DBD) and Updated Final Safety Analysis Report (UFSAR). Administrative controls have been put in place on the alignment restrictions for piping that could affect the operability of the RWST. Additional corrective actions planned include DBD and UFSAR changes and modifications of applicable plant procedures. The condition described in this Licensee Event Report is reportable in accordance with 10 CFR50.73(a)(2)(i)(B), any operation or condition which was prohibited by the plant's Technical Specifications. |
| | **2592011003**<br>Revision 0 | 12/21/2011<br>07/01/2011 | 05/02/2011 | Browns Ferry 1 | **Loss of Safety Function (SDC) Resulting from Emergency Diesel Generator Output Breaker Trip**<br>*Abstract:* On April 27, 2011, severe weather in the Tennessee Valley Service Area caused grid instability and loss of all 500-kV offsite power sources that resulted in scrams of all three Browns Ferry Nuclear Plant (BFN) units. On May 2, 2011, at approximately 0626 hours Central Daylight Time, with all three BFN units in cold shutdown and power supplied by onsite emergency diesel generators (EDGs) and one 161-kV offsite power source, the output breaker of the Unit 1/2 A EDG tripped. The "A" EDG output breaker tripped interrupting power to 4-kV Shutdown Board "A", causing a loss of power to a portion of the Unit 1 Reactor Protection System, and leading to Primary Containment Isolation System Group 2, 3, 6, and 8 isolations. The Group 2 isolation caused the loss of Shutdown Cooling on Unit 1 for approximately 57 minutes. Unit 2 was not affected by this event. Control room annunciation indicated an overspeed trip condition with the "A" EDG. The underlying cause for the "A" EDG output breaker trip was inadvertent (false) actuation of the overspeed trip limit switch. The root cause of this event was concluded to be inadequate technical rigor applied by Site Engineering personnel to recognize system vulnerabilities. |
| | **3052011002** | 05/03/2011 | 03/10/2011 | Kewaunee | **Loss of Station Backfeed Results in Loss of One Train of Offsite Power during Refueling Outage**<br>*Abstract:* At 1549 CST on March 10, 2011 with the plant shutdown and the reactor defueled, power was lost to safeguards 4160 volt bus 6. Emergency |

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| | | | | | diesel generator B started and re-energized bus 6. At the time of the event, bus 6 was energized from the main auxiliary transformer on backfeed. The event was caused by the opening of the 138kV breaker TA2066 as a result of an error by technicians working in the substation. All equipment operated as expected for the voltage restoration to safeguards bus 6. Safeguards bus 5 remained energized from offsite power through the tertiary auxiliary transformer during the event. Spent fuel pool cooling train A remained in operation during the event. Spent fuel cooling train B was restarted following restoration of power to bus 6. The event also caused loss of non-safeguards 4160 volt bus 4. In response to the loss of power to bus 4, the technical support center/station blackout diesel generator started but failed to load on 480 volt bus 46 resulting in continued loss of power to the technical support center. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A) for any event or condition that resulted in the automatic actuation of emergency ac electrical power systems. |
| | **4822011002** | 04/25/2011 | 02/22/2011 | Wolf Creek | **Diesel Generator Declared Inoperable Due to Inadequate Installation of a Fuel-Rack Control Pin** *Abstract:* On February 22, 2011, with the plant at 100 percent power in Mode 1 and the "A" Diesel Generator (DG) in stand-by condition, a Wolf Creek Nuclear Operating Corporation (WCNOC) engineer on a system walk down identified that a control pin on the fuel rack for the "A" DG was not completely inserted and not secured by a washer and cotter pin in accordance with the design. The "A" DG was declared inoperable at 1537 Central Standard Time (CST) on February 22, 2011 and returned to service at 0520 CST on February 23, 2011 after the control pin, washer and cotter pin were properly installed. The "B" DG and both offsite circuits were operable while the "A" DG was inoperable on February 22 and 23, 2011. The "A" DG may have been inoperable from 0200 CST on December 3, 2010, when it was removed from service for planned maintenance, to 0520 CST on February 23, 2011. Evaluations are in progress to determine the root cause of this event and the impact of an incorrectly installed fuel-rack control pin on diesel generator operation. |
| | **3902012001** | 03/16/2012 | 02/13/2011 | Watts Bar 1 | **Failure to Meet Technical Specifications due to Issues Associated with Vital Battery Surveillance Program** *Abstract:* On 01/17/2012, TVA determined that Vital Battery IV (VB4) was inoperable between 02/13/11 and 12/03/2011. This was based on an independent analysis of test data from the performance of Surveillance Requirement (SR) 3.8.4.14 for VB4 conducted on 02/10/2011 that indicated |

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| | | | | | the actual battery capacity did not meet the SR 3.8.4.14 acceptance criterion. On 11/21/2011, VB3 did not meet the SR 3.8.4.14 acceptance criterion. On 03/14/2012, TVA concluded that VB3 and VB4 may have been inoperable for unknown periods of time prior to the failed capacity tests. As a result, there were times when VB3 and VB4 were required to be operable to comply with Technical Specification (TS) 3.8.4 and TS 3.8.5, and WBN, Unit 1 failed to meet the applicable requirements of TS 3.8.4, TS 3.8.5 and LCO 3.0.4. Also, VB3 and VB4 may have been inoperable concurrently; thus, the requirements of LCO 3.0.3 may not have been met. During the time periods discussed above, VB3 and VB4 were capable of performing their safety function. Preliminarily, TVA determined that a manufacturing deficiency was the direct cause of the unexpected degradation of VB3 and VB4. The causes were inadequacies in the oversight of the battery surveillance program, and issues with the battery capacity test procedure. Corrective actions include changes to the battery test program, procedure revisions, and training of plant personnel. |
| | **4582011001** | 03/21/2011 | 01/20/2011 | River Bend | **Unplanned Actuation of Standby Service Water System due to Procedure Inadequacy**<br>*Abstract:* At 2:34 p.m. CST on January 20, 2011, while the plant was in a refueling outage, standby service water (SSW) pump "C" started automatically during system realignment. The Division 1 SSW subsystem (pumps "A" and "C") was being started to facilitate maintenance on the normal service water system. When the "A" pump was manually started, the pressure transient caused by the realignment of the motor-operated valves in the system caused a momentary low system pressure, actuating SSW pump "C" automatically. This event resulted from a weakness in the operating procedure, in that the intended system configuration for this operation exceeded the flow capacity for one pump. Actions are being taken to strengthen this and other similar procedures to prevent recurrence. This event is being reported in accordance with 10CFR50.73(a)(2)(iv)(A) as a condition that resulted in the automatic actuation of the "C" SSW pump. |
| | **4822011001**(C)* Cancellation Letter | 04/09/2011 | 01/03/2011 | Wolf Creek | **Potential for a CVCS Through-Weld Leak to Affect Reactor Coolant System Inventory After a Loss of Coolant Accident**<br>*Abstract:* At 1551 on January 3, 2011 with the plant at 100 percent power in Mode 1, operators identified a 300 drop-per-minute leak from a weld on a three-quarter-inch connection that joined a four-inch line in the Chemical Volume and Control System (CVCS) to a vent-valve assembly. A through-weld crack caused the leak and the weld was repaired on January 4, 2011. The vent-valve assembly was installed on October |

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| | | | | | 24, 2009 during Refueling Outage 17. Operations initiated a reportability evaluation on January 5, 2011 to determine whether this event should be reported as a condition prohibited by technical specifications under 10 CFR 50.73(a)(2)(i)(B). Wolf Creek Nuclear Operating Corporation (WCNOC) determined on March 26, 2011 that this event was not required to be reported under 10 CFR 50.73(a)(2)(i)(B) because this failure would not prevent the CVCS from fulfilling the high-pressure safety injection (HPSI) function. However, on April 1, 2011, based on questioning by a Nuclear Regulatory Commission inspector during an In-Service Inspection, it was determined that WCNOC did not address the extent to which the CVCS leak would impact Reactor Coolant System (RCS) inventory after a Loss of Coolant Accident. WCNOC is currently evaluating this issue. |
| | **3892011001** | 01/29/2011 | 12/02/2010 | St. Lucie 2 | **Inadvertent Crosstie of Component Cooling Water (CCW) to Control Room A/C Units** *Abstract:* On December 2, 2010 at 2320 St. Lucie Unit 2 was in Mode 1, while performing Plant Operator daily rounds, it was discovered the Unit 2 component cooling water (CCW) train A and B headers were cross-connected at control room A/C unit HVA/ACC-3C via CCW cooling flow being supplied from the A header and returning to the B header. This condition was a violation of Technical Specification (TS) 3.7.3 and required entry into a 1-hour shutdown action statement in accordance with TS 3.0.3. The Control Room staff immediately realigned CCW to 2-HVA/ACC-3C to Train B; no power reduction was necessary. This event is reportable as a condition prohibited by Technical Specifications, 10CFR50.73 (a) (2) (i) (B) and 10CFR50.73 (a) (2) (vii), a common cause inoperability of independent trains. A root cause evaluation (RCE) determined the valve misalignment relied on one procedure as the sole configuration control method for the CCW valves to the control room A/C Unit HVA/ACC-3C. Contributing causes included inadequate instructions for verifying CCW valve position to control room A/C unit HVA/ACC-3C by Procedure 2-0010123, "Administrative Control of Valves, Locks and Switches". Corrective actions included procedure revisions to ensure CCW and ICW valves have positions tracked when transferring the 2AB bus and components from A side to B side and from B side to A side and that locked open / locked closed tags reflect the current alignment of the valves. |
| | **3482010004** | 12/14/2010 | 10/29/2010 | Farley 1 | **Loss of Refueling Integrity** *Abstract:* On October 29, 2010 at 14:00, it was determined that Unit 1 was not in compliance with Technical Specification (TS) 3.9.3 in that one penetration providing direct access to the outside atmosphere was not isolated while Core Alterations were in progress. Prior to |