UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		February 13, 2012
.	)	

* * * * *

INTERVENORS' COMBINED REPLY IN SUPPORT OF MOTION FOR ADMISSION OF CONTENTION NO. 5

Now come Beyond Nuclear, Citizens Environment Alliance of Southwestern Ontario (CEA), Don't Waste Michigan, and the Green Party of Ohio (collectively, Intervenors), by and through counsel, and reply in support of their "Motion for Admission of Contention No. 5."

A. FENOC and the Staff are Equitably Estopped From Asserting Untimeliness By Their Individual And Collective Misrepresentations

FirstEnergy's position is that the cracking phenomenon in the Davis-Besse shield building first surfaced in October 2011, that 60 days had expired weeks, or days, before Intervenors' motion, and that time had run out, pure and simple. FENOC grants that its anticipated issuance by February 28, 2012 of a "root cause analysis" for the NRC might introduce additional facts for a renewed contention, but for the present, there is no permissible cracking contention because of the timeliness problem. FENOC's dogma is reminiscent of the "jam tomorrow" dilemma posed by the obstreperous White Queen in Through the Looking Glass; Intervenors are condemned forever to have filed either too early or too late, and as such, are consigned eternally to a

purgatory of untimeliness:[1]

> "I'm sure I'll take you with pleasure!" the Queen said. "Twopence a week and jam every other day."
> Alice couldn't help laughing, as she said "I don't want you to hire me — and I don't care for jam."
> "It's very good jam," said the Queen.
> "Well, I don't want any to-day, at any rate."
> "You couldn't have it if you did want it," the Queen said. "The rule is, jam to-morrow and jam yesterday - but never jam to-day."
> "It must come sometimes to 'jam to-day', Alice objected.
> "No, it can't," said the Queen. "It's jam every other day: to-day isn't any other day, you know."

But the dogma must be rejected. FENOC's position diverts attention from the utility's active concealment of the true nature of the cracking problem throughout the months of October through December 2011 by pretending that only the "decorative" and "architectural" features of the shield building were showing concrete fissures. Only on October 31, 2011, in a tersely-worded advisory to investerors did FirstEnergy guardedly admit that there was cracking in other than such areas of the shield building. And that advisory did not divulge all information which FirstEnergy knew on October 31. FENOC asserts in its Answer (at 44) that the utility "has reported cracking in areas not located near the maintenance cuts." A January 31, 2012 inspection report, ML12032A119,[2] shows that FENOC discovered on October 31, 2011 that there were other areas of cracking, but also:

> On October 31, 2011, the licensee identified additional indications of concrete cracking during IR testing towards the top of the SB wall, approximately between the 780 ft and 800 ft elevations. This area of indications was yet another one different from the laminar cracking initially identified adjacent to the RRVCH opening. The licensee

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[1]From Carroll, Lewis, Through the Looking-Glass, and What Alice Found There, 1871.

[2]http://adamswebsearch2.nrc.gov/webSearch2/doccontent.jsp?doc={99E65968-3B8D-471D-B9B9-65CDA18AE0CC}

> entered this extent-of-condition issue for the SB cracking into their CAP as CR 2011-04648, informed the NRC via the Resident Inspectors' Office on site, and continued to investigate further to determine if any additional adverse conditions existed.

P. 48 of attached report (p. 52 of .pdf). But FENOC did not publicly mention this discovery until January 5, 2012. It took Congressman Kucinich's public December 7, 2011 disclosures of the latest status of the cracking for the public to learn for the very first time that NRC "'impact response mapping' had revealed similar cracks in 'various areas of the top 20 feet of the building' that were not flute shoulders,"[3] and that this cracking "seems to be 'more extensive on the south side of the building.'" The NRC revealed "laminar cracking" that is "circumferential to the entire outer rebar map."[4] NRC staff further indicated to Congressman Kucinich that it was "assuming for purposes of evaluation that the flute shoulders have laminar cracking 'all the way up and down' the concrete wall"[5] - a thesis which Region III NRC personnel confirmed at the January 5, 2012 public presentation. In other words, the NRC instructed FENOC, in light of the revelation on October 31, 2011 of, among other things, the cracking within 20 feet of the summit of the shield building, to consider that cracking may be present throughout its entire 225-foot height. This stunning information was divulged neither to investors nor the public on October 31 by FENOC. FENOC also announced in the investor notice that it anticipated returning Davis-Besse to service within 30 days - the priority regulatory act it sought. And on October 31, 2011, FENOC was not yet facing the inconvenient prospect of a public meeting, where disclosure of the top-of-building cracking could cause public concern.

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[3] http://kucinich.house.gov/UploadedFiles/Factual_Analysis_of_FirstEnergy_Statements.pdf

[4] *Id.*

[5] *Id.*

Throughout October, and well into the month of November, 2011, the NRC reassured the public, leaving the strong impression that there would be no restart until the root cause of the cracking had been isolated, there was a grasp of how extensive the cracking was, emplacement of a corrective action plan and establishment of a monitoring system. The restart happened in early December 2011, apparently absent the accomplishment of any of these promised steps.

But the Intervenors and the public in general had a right to rely on the regulatory promises of the NRC and the supposed cooperation being displayed by FENOC prior to filing a new contention.

Even now, FENOC's minimal October 31 concession of "subsurface hairline cracks in two localized areas of the Shield Building similar to those found in the architectural elements" has not been supplemented by information FENOC possessed on the very same day - information which suggests a much more generalized cracking problem in the shield building. Imagine the difference in public understanding and perception of the problem had the November 1 media reportage stated, "FENOC Admits Possibility of Generalized Cracking Throughout Building," instead of "More Cracks Are Found in Davis-Besse Building," (which is how the Toledo Blade headlined its coverage). By controlling the nature and implications of information disclosed, FENOC may have averted a major disinvestment crisis, but the utility cannot similarly employ its mendacity to accuse Intervenors of missing the 60-day filing window.

If the fraudulent conduct of the defendant caused the injured party to remain ignorant of the violation, without any fault or lack of due diligence, the limitations period does not begin to run until the fraud is discovered. *Bailey v. Glover*, 88 U.S. (21 Wall.) 342 (1874); *Hobson v. Wilson*, 737 F.2d 1, 34 (D.C. Cir. 1984). The application of equitable principles is warranted

when a defendant fraudulently conceals its actions, misleading a plaintiff respecting the plaintiff's cause of action. See *School Dist. of Allentown v. Marshall*, 657 F.2d 16, 19-20 (3d Cir.1981); *Andrews v. Orr*, 851 F.2d 146, 151 (6th Cir.1988); *Dayco Corp. v. Goodyear Tire & Rubber Co.*, 523 F.2d 389, 394 (6th Cir.1975). Where a petitioner relied to its detriment on Staff's representations that no action would be immediately taken on licensee's application for renewal, elementary fairness requires that the action of the Staff could be asserted as an estoppel on the issue of timeliness of petition to intervene, and the petition must be considered even after the license has been issued. *Armed Forces Radiobiology Research Inst*. (Cobalt-60 Storage Facility), LBP-82-24, 15 NRC 652, 658 (1982), rev'd on other grounds, ALAB-682, 16 NRC 150 (1982).

FENOC and the NRC Staff are equitably estopped from benefitting from their repetitions in October and November 2011 that the cracking phenomenon was being regulatorily handled and no decision on reactor restart would happen in the absence of a root cause analysis - which as of this writing has yet to be provided the NRC by FENOC. Additionally, FENOC is equitably estopped from claiming the 60-day window had closed, because of its own incomplete public information disclosures.

B. The Contention Was Timely Filed Based On Available Facts

1. A Party Is Not Required To 'Piece Together Disparate Shreds of Information', But Intervenors Attempted to Do So Nonetheless

The ASLB in the Prairie Island relicensing proceeding stated, "we are 'not impressed with arguments suggesting that, in order to raise a timely contention, a party must piece together disparate shreds of information that, standing alone, have little apparent significance.' The *Prairie Island* Board rejected such an argument, explaining:

> Applicant and NRC Staff . . . claim that [the Petitioner] could have filed this

> contention in the wake of numerous events that transpired since late 2008. Those events include meetings and reports on the cavity leakage issue, the issuance of yellow and white findings in early 2009, NSPM's responses to the NRC Staff's requests for additional information (RAIs), and issuance of the preliminary SER in June 2009, which identified the cavity leakage issue as an open item. All of those events revealed fragments of the story that PIIC would ultimately fashion into a contention. But none of those events, by itself, fully captured the scope of PIIC's concerns related to safety culture. We would not expect PIIC to 'piece together' those 'shreds of information' and formulate its contention prior to issuance of the final SER. Accordingly, we find that PIIC's contention meets all of the criteria at 10 C.F.R. § 2.309(f)(2).

Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), Order (Narrowing and Admitting PIIC's Safety Culture Contention) (Jan. 28, 2010) at 6-7 (unpublished) (quoting *U.S. Dep't of Energy* (High Level Waste Repository), LBP-09-29, 70 NRC __, __ (slip op. at 12) (Dec. 9, 2009). Despite Staff and FENOC prevarications which afforded Intervenors only a limited perspective on the scope and locations of the cracking over a two-month period, Intervenors did their best to compile together various shreds of information to formulate their contention. And at that, FENOC challenges the legitimacy of the contention for *Intervenors*' lack of a root cause analysis. Even under the new standard pronounced by the full Commission when it reversed the *Prairie Island* ASLB, Intervenors have certainly not "delay[ed] filing a contention until a document becomes available that collects, summarizes and places into context the facts supporting that contention. " *Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), CLI-10-27 at 17. The significant facts of the cracking within 20 feet of the top of the shield building, and the NRC's 225-foot-long scope of potential cracking, became publicly known only on December 7, 2011, following a December 6 meeting between Congressman Kucinich and NRC staff (the day after restart of the reactor), after which the congressman publicly circulated the new revelations. That information, and the date of its disclosure, were cited by the Intervenors in their Motion (at pp. 32-35). Intervenors brought their

motion 34 days after learning of the latest significant facts (and after hearing the upper 20-foot cracking and 225-foot length assumption affirmed at the NRC's public meeting on January 5, 2012). Intervenors were well within the window for filing.

C. Reply As To 'Inappropriate Broadening' of Proceedings

FENOC argues (Answer at 18) that the new contention raises a safety issue which is "a completely separate topic" from the two pending NEPA contentions, and would "inappropriately broaden the current proceeding and could result in corresponding delay." Apart from the weak concept of opposition for want of a topical relationship to existing contentions, concerns about "inappropriate broadening" should be junior to the as-yet undiagnosed problem affecting a key safety structure in the generating complex. The NRC staff has suggested that the cracking is an age-related problem in the wording it chose to rephrase intervenors' cracking contention,[6] wherein the Staff expressed concern that absent an aging management plan, the shield building might be "unable to perform its intended functions of: 1) protecting the steel containment from environmental effects, including wind, tornado, and external missiles, 2) providing biological shielding, 3) providing controlled release to the annulus during an accident, and 4) providing a means for collection and filtration of fission product leakage from the Containment Vessel following a hypothetical accident." NRC Staff Answer at 2.

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[6]Proposed reworded contention, from Staff Answer at 2:

Is the Structures AMP adequate to address any aging effects for the shield building that are related to the cracks identified by FENOC during the October 10, 2011 reactor head replacement and subject to a root cause evaluation to be provided by FENOC on February 28, 2012 such that the shield building would be unable to perform its intended functions of: 1) protecting the steel containment from environmental effects, including wind, tornado, and external missiles, 2) providing biological shielding, 3) providing controlled release to the annulus during an accident, and 4) providing a means for collection and filtration of fission product leakage from the Containment Vessel following a hypothetical accident?

Hence according to the NRC, the cracking contention would "appropriately," not "inappropriately," broaden the proceedings. The contention raises NEPA as well as safety considerations, mirroring what the shield building itself is designed to address (*viz*., protection of the environment as well as public health and safety).

D. Intervenors Question ER Adequacy in Light of Shield Building Cracks

The NRC Staff (Answer at 29), asserts that "assuming that Intervenors are not challenging the environmental impacts of the cracks themselves, and instead seek to challenge the ER's discussion of the environmental impacts of any repairs or refurbishment of the shield building, that challenge is simply premature and speculative." But the Intervenors ***are*** questioning the adequacy of the ER in light of the environmental impacts of the shield building cracks, because the ER mentions no cracking, nor does it provide analysis of the potential environmental impacts of the cracks that are being identified. In light of the Staff's assertion, *id.*, that "no repairs or refurbishment is currently contemplated" and that "it would be too speculative for FENOC to discuss the environmental impacts from an unknown repair or refurbishment that may or may not occur," Intervenors' motion is very timely. It identifies a new and, presently at least, continuing deficiency in that the ER omits to identify and discuss the cracks and their environmental implications.

Oddly, the Staff accuses Intervenors of having "proffered no expert, let alone an expert opinion sufficient to tie the cracks in the shield building to an environmental impact." Staff Answer at 29. But the Staff itself posited in a Request for Additional Information (RAI) to FENOC that "[e]xtensive cracking in the shield building could affect the structural integrity of the shield building and may impact its ability to perform its intended function during the period

of extended operation." NRC RAI B.2.39-13, cited at Staff Answer 3. And, as well, there is a judicial admission implicit in the reworded contention the Staff has proferred in its Answer, see fn. 6 *infra*.

<u>E. The Cracking Problem Could Allow a Challenge To the Category 1 Determination for Postulated Accidents</u>

FENOC maintains that "Intervenors challenge the Commission's 'Category 1' determination for 'Postulated Accidents' in 10 C.F.R. Part 51, Subpart A, Appendix B based on the Shield Building cracking" (FENOC Answer at 25) and that it is an impermissible challenge to the regulations. But implicit in the Category 1 determination for Davis-Besse is the assumption of an ***uncracked*** shield building. The regulatory scheme does not contemplate a flat prohibition on raising Category 1 issues; they may be raised when a petitioner demonstrates that "there is new and significant information subsequent to the preparation of the GEIS regarding the environmental impacts of license renewal." *Matter of Florida Power & Light Co.* (Turkey Point Nuclear Power Plant), CLI-01-17, 54 NRC 10-12 (7/19/2001); *see also* 10 C.F.R. § 51.53(c)(3)(iv) (new and significant information). Only by ignoring the wholly-unique cracking phenomenon at Davis-Besse can FENOC make such generalizations as (Answer p. 28) that Intervenors could not show a "connection between Shield Building cracking and consideration of design basis accidents during the license renewal environmental review." Only by ignoring the unique cracking problem (*id.* at 29) can FirstEnergy try to bar Intervenors from confronting the regulatory dogma "that, for any license renewal of a nuclear power plant, the probability-weighted consequences of a severe accident are small." Surely the shield building cracking at Davis-Besse, which is distinguishable from the only other known crumbling shield building, at Crystal River in Florida, likely meets the "unusual and compelling circumstances" standard of 10

C.F.R. §2.335.

FENOC's serial trivializations of the unprecedented problems at Davis-Besse reveal, overall, that the License Renewal Application has not yet caught up to the reality being documented at the plant. What FENOC insultingly calls Intervenors' "environmental musings" are fact-based suggestions that serious reconsideration of LRA adequacy and management of the environmental and safety implications of Davis-Besse cracking must be found acceptable before the deteriorated carbonating concrete of the shield building can be deemed adequate for an additional score of years.

FENOC's profound insincerity on this point is most apparent in its Answer at 35-36, where it summarizes factual arguments of Intervenors which, to FENOC, are "neither germane to age-related degradation nor unique to the period covered by the . . . license renewal application."[7] But there is a reasonably close causal relationship between the Davis-Besse relicensing and the need for adequate structural integrity of the shield building in the 2017-2037 time period, and analysis of that relationship must account for actual and anticipated physical changes and exterior environmental factors occurring both before and after 2017. Those changes include whatever

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[7]"• Complaints about the NRC authorizing restart of Davis-Besse and the sufficiency of the NRC's or FENOC's assessments of the Shield Building cracking in determining to restart the plant;

• Complaints about FENOC's planned steam generator replacement at Davis-Besse in 2014 (prior to the period of extended operation);

• Complaints about the evolving knowledge of the extent of the Shield Building cracking, and FENOC's and the NRC's reporting of the extent of cracking;

• Complaints about the current safety (i.e., prior to the period of extended operation) of the plant due to the Shield Building cracking;

• Complaints about the adequacy of repairs, including those to the Shield Building and the inner steel containment, due to activities conducted prior to the period of extended operation;

• Complaints about the NRC's review of the Shield Building cracking, including the timing and content of its inspector's requests for information from FENOC; and

• Complaints about seismic hazards at the site."

repairs might be contemplated in the coming months and years, the planned reopening of the shield building in 2014 for steam generator replacement, and perhaps other foreseeable physical changes to, and repairs of, the structure.

The Supreme Court has twice addressed the causal relationship which must exist in order to invoke NEPA. One decision addressed the resumption of power generation activity at the Three Mile Island nuclear power plant after the infamous 1979 accident. *Metro. Edison Co. v. People Against Nuclear Energy*, 460 U.S. 766, 768, 103 S.Ct. 1556, 75 L.Ed.2d 534 (1983). Though officially, supposedly insignificant amounts of radiation were assumed to have been released in the accident, the partial meltdown and contained hydrogen explosion that occurred caused widespread concern about the safety of the plant. People Against Nuclear Energy (PANE) argued that restarting the reactor would "cause both severe psychological health damage to persons living in the vicinity[ ] and serious damage to the stability, cohesiveness, and well-being of the neighborhood communities." *Id.* The NRC declined to take evidence on this issue, and PANE petitioned for review, arguing that both NEPA and the AEA required such an analysis. *Id.* at 770, 103 S.Ct. 1556. The D.C. Circuit agreed as to NEPA, but the Supreme Court reversed. The Court noted that, in order to determine when NEPA requires consideration of a particular environmental effect, agencies and reviewing courts "must look at the relationship between that effect and the change in the physical environment caused by the major federal action at issue." *Id.* at 773, 103 S.Ct. 1556. The Court held that NEPA attaches only when there is a "reasonably close causal relationship between a change in the physical environment and the effect at issue." The Court likened this relationship to "the familiar doctrine of proximate cause from tort law." *Id*. at 774, 103 S.Ct. 1556. In the case before it, the Court observed that the renewed operation of

the reactor would affect the environment, particularly in the release of low levels of radiation, increased fog, the release of warm water into the Susquehanna River, and the potential results of a nuclear accident. *Id.* at 775, 103 S.Ct. 1556.

The Supreme Court again discussed the NEPA causation requirement in *Department of Transportation v. Public Citizen*, 541 U.S. 752, 124 S.Ct. 2204, 159 L.Ed.2d 60 (2004). *Public Citizen* concerned the operation of Mexican tractor-trailer trucks in the United States and the scope of an environmental assessment performed on new regulations which would allow them access to the U.S. The EA did not consider the environmental impact of increased Mexican truck traffic because the Federal Motor Carrier Safety Administration attributed this increase not to the regulations but to NAFTA and the President's decision to lift the moratorium. *Id.* at 761, 124 S.Ct. 2204. The Supreme Court upheld the FMCSA's decision, noting that an EIS is required only for "'major Federal actions,'" defined to include "'actions with effects that may be major and which are potentially subject to Federal control and responsibility.'" *Id*. at 763, 124 S.Ct. 2204 (quoting 40 C.F.R. §1508.18). The Court then noted that "effects" were limited by regulation to (1) "[d]irect effects, which are caused by the action and occur at the same time and place," and (2) "indirect effects, which are caused by the action and are later in time or farther removed in distance, but are still reasonably foreseeable." *Id.* at 764, 124 S.Ct. 2204 (internal quotation marks and citation omitted).

In considering the causal relationship between the agency action and the environmental impact, as required by *Metropolitan Edison*, the Court found that "'but for' causation, where an agency's action is considered a cause of an environmental effect even when the agency has no authority to prevent the effect," is "insufficient to make an agency responsible for a particular

effect under NEPA." *Id*. at 767, 103 S.Ct. 1556. The Court ruled that "the legally relevant cause of the entry of the Mexican trucks is not FMCSA's action, but instead the actions of the President in lifting the moratorium and those of Congress in granting the President this authority." *Id.* at 769, 124 S.Ct. 2204 and directed that the agency may "draw a manageable line between those causal changes that may make an actor responsible for an effect and those that do not." *Id.* at 767, 124 S.Ct. 2204 (quoting *Metro. Edison*, 460 U.S. at 774 n. 7, 103 S.Ct. 1556). In the cases, this line appears to approximate the limits of an agency's area of control. Since here the NRC can control Davis-Besse and its operation, consideration under NEPA of the aging management of the shield building is easily obligatory.

F. Cumulative Effects Analysis Requires Consideration of Changes To the Shield Building During the Period 2012-2017

FENOC argues to deny the legitimacy of NEPA cumulative effects analysis of events that might impinge upon the adequacy of aging management of the shield building, such as the planned 2014 cut into the shield building mentioned by the Intervenors, seismic activity in the vicinity of Davis-Besse, or physical changes brought on by the very repairs aimed at rectifying the cracking. FENOC in effect argues that cumulative effects analysis is somehow unnecessary here, and FENOC is wrong.

The CEQ's regulations define a project's cumulative impacts as “the impact on the environment which results from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions regardless of what agency (Federal or non-Federal) or person undertakes such other actions.” 40 C.F.R. §1508.7; see also 40 C.F.R. §1508.25 (requiring that agencies take cumulative impacts into consideration during NEPA review). The regulation states that “[c]umulative impacts can result from individually minor but

collectively significant actions taking place over a period of time." 40 C.F.R. §1508.7. A consideration of cumulative impacts must also consider "[c]losely related and proposed or reasonably foreseeable actions that are related by timing or geography." *Vieux Carre Prop. Owners, Residents, & Assocs., Inc. v. Pierce*, 719 F.2d 1272, 1277 (5th Cir.1983). "[R]easonably foreseeable action[s], for which cumulative impacts must be analyzed . . . include proposed actions." *N. Alaska Envtl. Ctr. v. Kempthorne*, 457 F.3d 969, 980 (9th Cir.2006) (internal quotation marks omitted). An action is not too speculative to qualify as a proposed action when the agency issues a notice of intent to prepare an EIS. *Id.*; accord, *City of Tenakee Springs v. Clough*, 915 F.2d 1308, 1313 (9th Cir.1990).

Rather than deny the potentiality and effects of physical changes to the shield building in the coming five years, FENOC must abide by NEPA's command that such cumulative effects be considered. FENOC chose the time to file its application, and in so doing, accepted the likelihood (indeed, inevitability) that maintenance and operations justifications would require one or more additional cuts through the shield building wall; that there might be seismic activity which could worsen the existing or repaired condition of the shield building; and that anticipated repairs might have implications for the future adequacy of the structure during the 20-year extension period.

Conclusion: Contention 5 Should Be Admitted for Hearing

Notwithstanding the arguments made by FENOC and the Staff that Intervenors must produce expert testimony, must display proof beyond a reasonable doubt of the environmental implications of cracking, or provide a root cause analysis to justify Intervenors' insistence that the shield building problems are of concern in this license extension case, a petitioner does not

have to prove its contentions at the admissibility stage. *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-04-22, 60 NRC 125, 139 (2004). The factual support required is "a minimal showing that material facts are in dispute." All that is needed at this juncture is "alleged facts" and the factual support "need not be in affidavit or formal evidentiary form and need not be of the quality necessary to withstand a summary disposition motion." *First Energy Nuclear Operating Company* (Davis-Besse Nuclear Power Station, Unit 1), ASLBP No. 11-907-01-LR-BD01, LBP-11-13 at 17 (April 26, 2011) (slip op.).

Intervenors have met, and exceeded, the evidentiary threshold for consideration of the Davis-Besse shield building cracking within this litigation, and within the ER. The cracking contention should be admitted for hearing.

WHEREFORE, Intervenors pray the Licensing Board admit Contention 5 to these proceedings.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		February 13. 2012
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CERTIFICATE OF SERVICE OF
INTERVENORS' COMBINED REPLY IN SUPPORT OF MOTION
FOR ADMISSION OF CONTENTION NO. 5

We hereby certify that a copy of the "INTERVENORS' COMBINED REPLY IN SUPPORT OF MOTION FOR ADMISSION OF CONTENTION NO. 5" was sent by us to the following persons via electronic deposit filing with the Commission's EIE system on the 13th day of February, 2012:

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January 31, 2012

Mr. Barry Allen
FirstEnergy Nuclear Operating Company
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SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION INTEGRATED INSPECTION REPORT 05000346/2011005

Dear Mr. Allen:

On December 31, 2011, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Davis-Besse Nuclear Power Station. The enclosed report documents the results of this inspection, which were discussed on January 10, 2012, with the Director of Site Operations, Mr. Brian Boles, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, two NRC-identified and four self-revealed findings of very low safety significance were identified. Four of these findings were determined to also involve violations of NRC requirements. In addition, one Severity Level IV violation was also identified by the NRC. However, because of the very low safety significance and because these issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs), in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any finding or NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspectors' Office at the Davis-Besse Nuclear Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-346
License No. NPF-3

Enclosure: Inspection Report 05000346/2011005
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2011005
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	October 1, 2011, through December 31, 2011
Inspectors:	D. Kimble, Senior Resident Inspector A. Wilson, Resident Inspector T. Briley, Reactor Engineer P. Cardona-Morales, Reactor Engineer T. Go, Radiation Protection Inspector M. Holmberg, Senior Reactor Inspector L. Jones, Jr., Reactor Inspector G. O'Dwyer, Reactor Inspector M. Mitchell, Radiation Protection Inspector J. Neurauter, Senior Reactor Inspector P. Smagacz, Reactor Engineer J. Steffes, Reactor Engineer
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

Inspection Report 05000346/2011005; 10/1/2011-12/31/2011; Davis-Besse Nuclear Power Station; Inservice Inspection Activities; Maintenance Risk Assessments and Emergent Work Control; Operability Evaluations; Post-Maintenance Testing; Outage Activities; Follow-Up of Events and Notices of Enforcement Discretion; and Other Activities.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Six Green findings and one Severity Level (SL) IV violation were identified by the inspectors. Four of the findings, as well as the SL IV violation, were dispositioned as non-cited violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a SL after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A self-revealed finding of very low safety significance was identified for the licensee's failure to establish, implement, and maintain technically adequate procedures to permit the proper switching of feedwater sources for the station's auxiliary boiler, such that when the switching of feedwater sources from demineralized water to the station's normal condensate system took place per approved procedures, there were detrimental results. Specifically, the approved procedures for this activity relied upon a check valve to keep the demineralized water header from being exposed to greater pressure than its design. When that check valve failed to function as designed, failure of demineralized water system components and the inadvertent deluge and failure of safety-related electrical equipment resulted.

  The finding was determined to be of more than minor significance because it was associated with the Initiating Events Cornerstone attribute of procedure quality and had adversely affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, electrical power to an entire string of safety-related 480 Vac motor control center (MCCs) (i.e., E11A, E11B, E11C, E11D, and E11E) was forced to be interrupted when a deficient procedure for the operation of the station's auxiliary heating boiler caused a significant amount of water to be deluged onto MCC E11C, resulting in an electrical short and fire within the MCC. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Because the finding involved reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process – Phase 1 Operational Checklists for Both PWRs and BWRs." Since the finding was associated with an issue that occurred during the time the licensee was in a cold shutdown (Mode 5) condition, the inspectors consulted Checklist 3, "PWR Cold Shutdown and Refueling Operation: Reactor Coolant System (RCS) Open and Refueling Cavity Level Less Than 23 Feet or

RCS Closed and No Inventory in the Pressurizer; Time to Boiling Less Than 2 Hours." The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes, nor did it meet any of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program (CAP) component, because the licensee did not take appropriate corrective actions to address the safety issue in a timely manner, commensurate with the safety significance and complexity. Specifically, the licensee had multiple previous opportunities to have appropriately diagnosed and corrected the issue, but failed to do so. (P.1(d)) (Section 4OA3.2)

- Green. A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," were identified by the inspectors for the licensee's failure to perform an adequate review of fabrication records to ensure material procured from a contractor (replaced reactor vessel closure head) met the construction code (CC). Specifically, the accessible surfaces of the 60 closure head flange stud holes were not subjected to dye penetrant or magnetic particle examinations as required by the CC. As a corrective action, the licensee completed magnetic particle examination of the accessible surfaces of the 60 flange stud holes prior to placing the vessel head into service.

  The finding was determined to be more than minor because it was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Absent NRC identification, the licensee would not have completed surface examination of the 60 flange stud holes to ensure unacceptable material flaws (e.g., cracks) were not placed in service. Because material flaws such as cracks serve as stress risers that reduce the ability of the replacement reactor vessel closure head to withstand failure by crack propagation during design basis events (e.g., pressurized thermal shock), they would place the reactor coolant system at an increased risk for through-wall leakage and/or failure. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Initiating Events Cornerstone. Because this finding was identified prior to placing the replacement reactor vessel closure head in service and no fabrication flaws were identified, the inspectors answered "no" to the SDP Phase 1 screening question "Assuming worst case degradation, would the finding result in exceeding the Technical Specification (TS) limit for any reactor coolant system leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?" Therefore, the finding screened as having very low safety significance. This finding had a cross-cutting aspect in the area of Human Performance, Decision Making because the licensee staff failed to demonstrate that nuclear safety was an overriding priority in decisions affecting the replacement reactor vessel closure head. Specifically, the failure to perform an adequate review of the replacement reactor vessel closure head fabrication records was caused by the licensee's decision to not review the manufacturer's interpretations and application of the CC rules. (H.1(b)) (Section 4OA5.3).

Cornerstone: Mitigating Systems

- Green. A self-revealed finding of very low safety significance (Green) was identified when low pressure injection equipment was damaged by operators attempting to access an overhead valve. Specifically, by climbing and standing on sensitive plant equipment, the licensee failed to comply with the standards and expectations for accessing plant equipment contained in procedure NOP-OP-1002, "Conduct of Operations". An immediate corrective action was taken to repair the damaged temperature element and restore low pressure injection pump no. 1 to operable status. A long-term solution to providing access to the overhead valve is under evaluation in the corrective action program.

  The inspectors determined that the finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Human Performance and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the damage caused when falling from plant equipment rendered low pressure injection train 1 inoperable. The inspectors evaluated the finding using IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, using the Phase 1 SDP worksheet for the Mitigating Systems Cornerstone. The finding screened as very low safety significance because the inspectors answered "no" to the screening questions in Table 4a. Specifically, the finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent actual loss of safety function of a single train for greater than its TS allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a cross-cutting aspect in the area of Human Performance, Work Control Component, because the licensee did not plan and coordinate work activities consistent with nuclear safety. Specifically, the licensee did not appropriately plan for job site conditions impacting human performance since an appropriate available method for accessing CC258 was not evaluated. (H.3(a)) (Section 1R13.1)

- Green. A finding of very low safety significance and an associated NCV of TS 5.4.1(a) were identified by the inspectors for the licensee's failure to establish, implement, and maintain technically adequate procedures to cover the restoration (i.e., filling and venting) of the component cooling water (CCW) system following maintenance activities. Specifically, a complex series of fill and venting evolutions to restore the system had been required following extensive maintenance activities; these evolutions did not ensure that all the air was vented from the system, such that later ultrasonic testing performed by the licensee identified a significant air void, approximately 19 cubic feet, in a CCW pump 3 horizontal suction piping segment. The issue was entered into the licensee's CAP as CRs 2011-05542 and 2011-05831.

  The finding was determined to be of more than minor safety significance because the issue was associated with the Mitigating Systems Cornerstone attribute of procedure quality, and had adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, CCW, a mitigating system, had its reliability adversely impacted by the lack of appropriate fill and venting procedural guidance. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." Because the finding involved

reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process - Phase 1 Operational Checklists for Both PWRs and BWRs.” Since the finding was associated with an issue that occurred during the time the licensee was conducting RCS fill and venting activities and plant conditions were in transition, the inspectors consulted both Checklist 2, “PWR Cold Shutdown Operation: RCS Closed and Steam Generators Available for Decay Heat Removal (Loops Filled and Inventory in the Pressurizer); Time to Boiling Less Than 2 Hours,” and Checklist 3, “PWR Cold Shutdown and Refueling Operation: RCS Open and Refueling Cavity Level Less Than 23 Feet or RCS Closed and No Inventory in the Pressurizer; Time to Boiling Less Than 2 Hours.” The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes on either checklist, nor did it meet any of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance. This finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee’s procedures and guidance for the restoration of the CCW system following outage maintenance activities did not ensure that the system was fully filled and properly vented prior to operation. (H.2(c)) (Section 1R15.1)

- <u>Severity Level IV</u>. The inspectors identified a SL IV NCV of 10 CFR 54(i) when a non-licensed member of the licensee’s engineering staff was observed operating switches that directly caused the insertion of various control rods that were being subjected to timing tests. Specifically, the inspectors observed that key switches used to interrupt power to the control rod drives and cause control rod insertion were manipulated by a member of the licensee’s engineering staff, and not a licensed individual. The issue was entered into the licensee’s CAP as CR 2011-06318.

  The issue was determined to be associated with the Mitigating Systems Cornerstone attribute of procedure quality. However, the inspectors subsequently determined that the issue had not adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because of several factors, the inspectors determined that the issue was of minor safety significance and, as such, did not constitute a finding. These factors included:

  - All control rod group withdrawal activities were accomplished from the control room by an on-watch licensed reactor operator;
  - All activities in the electrical penetration room were performed in accordance with an approved written test procedure, and under the direct supervision of a licensed Senior Reactor Operator;
  - The operation of the local key switches in the electrical penetration room, albeit by a non-licensed individual, could only cause control rod insertion. There was no withdrawal capability; and
  - The individual operating the local key switches in the electrical penetration room was always in continuous communication with the on-watch licensed reactor operator in the control room.

  The inspectors determined that the issue was subject to the NRC’s traditional enforcement process as an issue that had the potential to impact the agency’s ability to

perform its regulatory function. Specifically, the NRC's Reactor Oversight Process fundamentally assumes that only duly licensed individuals are allowed to manipulate reactor controls and alter core reactivity or make changes to reactor power, and that all licensed individuals perform their licensed duties in accordance with any restrictions associated with their individual licenses. The inspectors conferred with NRC Region III management and members of the enforcement staff and determined that, because of the factors noted above, the issue constituted a SL IV violation that resulted in no, or relatively inappreciable, safety consequences. Because this issue was dispositioned through the traditional enforcement process and had no Reactor Oversight Process aspects, there was no cross-cutting aspect associated with the violation.
(Section 1R19.1)

- Green. A finding of very low safety significance and an associated NCV of TS 5.4.1(a) were identified by the inspectors for the licensee's failure to establish, implement, and maintain technically adequate procedures and drawings to cover the restoration (i.e., motor controller re-energization) of components in the CCW system following maintenance activities. Specifically, as circuit breaker BE1161 was closed to restore power to motor-operated valve (MOV) CC2645, the train 1 auxiliary building return header isolation valve, the MOV unexpectedly stroked open resulting in a rapid loss of CCW system inventory and a low level alarm for the CCW surge tank. Subsequent investigation revealed that notes describing the operating logic for CC2645 on approved operational drawings were less than adequate. The issue was entered into the licensee's CAP as CR 2011-04078.

  The finding was determined to be of more than minor safety significance because the issue was associated with the Mitigating Systems Cornerstone attribute of procedure quality, and had adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, CCW, a mitigating system, had its reliability adversely impacted by the inadequate procedural guidance for motor controller restoration. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." Because the finding involved reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process – Phase 1 Operational Checklists for Both PWRs and BWRs." Since the finding was associated with an issue that occurred during the time the reactor was in a defueled condition, the inspectors conservatively consulted all four PWR checklists (i.e., Checklists 1 – 4). The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes on any checklist, nor did it meet any of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance. This finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee's procedures, drawings and guidance for the restoration of the CCW system following outage maintenance activities did not ensure that the system was properly aligned prior to restoration of electrical power to MOV CC2645. (H.2(c))
  (Section 1R20.1)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" were identified by the inspectors for the licensee's failure to control weld rod oven temperature in accordance with procedure WFMC-1 during a rebar splice weld completed for restoration of the shield building access opening. As a corrective action, the licensee removed the welder's certification to weld rebar and documented this issue in CR 2011-05536. To ensure that the horizontal rebar splice weld 2H-03R was not affected by delayed hydrogen cracking, the licensee's vendor examined the weld splice 48 hours after fabrication and did not identify cracks.

  The finding was determined to be more than minor because the finding was associated with the Barrier Integrity Cornerstone attribute of Configuration Control and adversely affected the cornerstone objective to provide reasonable assurance that the physical design barriers (e.g., containment) protect the public from radionuclide releases caused by accidents or events. The shield building is part of the containment system. Absent NRC identification, rebar welds would have been fabricated with electrodes exposed to ambient temperatures for excessive periods of time creating a condition that results in hydrogen-induced weld cracking. Rebar splice material with cracks returned to service would increase risk for shield building failure during design basis events such as wind-driven missile impact or earthquake-induced loads. The inspectors completed a significance determination, in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Containment Barrier. Because the issue was corrected promptly, prior to introduction of weld material with hydrogen-induced cracks, the inspectors answered "no" to each of the four Phase 1 screening questions. Therefore, the finding screened as having very low safety significance. This finding had a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the failure to control the weld rod oven temperature in accordance with procedure WFMC-1 was caused by inadequate licensee oversight of the contracted welder. (H.4(c)) (Section 1R08.1).

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

Summary of Plant Status

At midnight on September 30/October 1, 2011, the main generator output breakers were opened and the unit was taken offline for mid-cycle outage 17M to facilitate replacement of the reactor vessel closure head. On December 5, 2011, the reactor was restarted and criticality achieved. The unit was synchronized to the main electrical grid and the main generator output breakers were closed on December 6, 2011. The unit reached full power operation 2 days later, on December 8, 2011, and remained operating at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed Corrective Action Program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- Ultimate heat sink; and
- Borated water storage tank and associated piping.

This inspection constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Alignment Verifications

a. Inspection Scope

The inspectors performed partial system alignment verifications of the following risk-significant systems:

- Service water train 1 in Mode 5 when lined up to support shutdown operations during the week ending November 19, 2011; and
- Emergency diesel generator (EDG) no. 2 when EDG no. 1 was unavailable for planned testing during the week ending December 24, 2011.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports (CRs), and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted two partial system alignment verification samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Alignment Verification

a. Inspection Scope

During the week ending December 10, 2011, the inspectors performed a complete system alignment inspection of the decay heat/low pressure injection system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was

performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system alignment verification sample as defined in IP 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Quarterly Tours

a. Inspection Scope

The inspectors conducted fire protection inspection tours which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Containment Elevation 565' (Room 217, Fire Area D);
- Containment Elevation 603' (Rooms 407 and 410, Fire Area D);
- Containment Elevation 636' (Room 580, Fire Area D); and
- Containment Elevation 643' (Rooms 700 and 701, Fire Area D).

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events (IPEEE) with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Heat Sink Performance Review

a. Inspection Scope

The inspectors reviewed the licensee's testing of the spent fuel pool heat exchangers to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

This annual heat sink performance review constituted a single inspection sample as defined in IP 71111.07-05.

b. Findings

No findings were identified.

.2 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, associated calculations, performance test results and cooler inspection results associated with the Component Cooling Water (CCW) heat exchanger number E22-3. This heat exchanger was chosen based on its risk significance in the licensee's probabilistic safety analysis, its important safety-related mitigating system support functions, its operating history, and its relatively low margin.

The inspectors verified that testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying the test method used was consistent with accepted industry practices, or equivalent, the test conditions were consistent with the selected methodology, the test acceptance criteria were consistent with the design basis values, and results of heat exchanger performance testing. The inspectors also verified that the test results appropriately considered differences between testing conditions and design conditions, the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values and test results considered test instrument inaccuracies and differences.

The inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors verified the methods used to inspect and clean heat exchangers were consistent with as-found conditions identified and expected degradation trends and industry standards, the licensee's inspection and cleaning activities had established acceptance criteria consistent with industry standards, and the

as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

In addition, the inspectors verified the condition and operation of the CCW heat exchanger number E22-3 were consistent with design assumptions in heat transfer calculations and as described in the final safety analysis report. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow-induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors verified the performance of ultimate heat sinks (UHS) and safety-related service water systems and their subcomponents such as piping, intake screens, pumps, valves, etc. by tests or other equivalent methods to ensure availability and accessibility to the inplant cooling water systems.

The inspectors reviewed the results of the licensee's inspection of the UHS weirs or excavations. The inspectors verified that identified settlement or movement indicating loss of structural integrity and/or capacity was appropriately evaluated and dispositioned by the licensee. In addition, the inspectors verified the licensee ensured sufficient reservoir capacity by trending and removing debris or sediment buildup in the UHS.

The inspectors reviewed the licensee's operation of service water system and UHS. This included the review of licensee's procedures for a loss of the service water system or UHS and the verification that instrumentation, which is relied upon for decision making, was available and functional. In addition, the inspectors verified that macrofouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors verified that licensee's biocide treatments for biotic control were adequately conducted and the results monitored, trended, and evaluated. The inspectors also reviewed strong pump-weak pump interaction and design changes to the service water system and the UHS. The inspectors also verified that the licensee maintained adequate pH, calcium hardness, etc.

In addition, the inspectors reviewed condition reports related to the heat exchangers/coolers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities constituted two heat sink inspection samples as defined in IP 71111.07 05.

b. Findings

No findings of significance were identified. Section 4OA2 documents a review of the licensee's assessment of the as-found condition of the intake canal.

1R08 Inservice Inspection Activities (71111.08P)

From October 11, 2011, through November 23, 2011, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), steam generator (SG) tubes, emergency feedwater (FW) systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, R08.3, IR08.4, and 1R08.5 below were completed in accordance with IP 71111.08. A full inspection sample was not available during this outage, so the reviews under Section 1R08.4 are considered incomplete. Additional reviews to complete this procedure will be documented in a future inspection report. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed the following non-destructive examinations mandated by the American Society for Mechanical Engineers (ASME) Code Section XI to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic examination (UT) of the pressurizer nozzle-to-lower head weld (RC-PZR-WP-15);
- Dye penetrant (PT) examination of pipe-to-valve weld (MU-31-CCA-18-1-FW23);
- UT of reactor vessel shell-to-lower head weld no. 4.; and
- UT of reactor vessel nozzle-to-shell weld no. 11.

The inspectors reviewed the following examination records (volumetric or surface) with recordable indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC approved alternative:

- PT examination report no. 17-PT-011, valve HP92- to-pipe weld.

The inspectors observed the following welds completed for risk significant systems during the outage to determine if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the construction code (CC). Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedures were qualified in accordance with the requirements of CC, the ASME Code Section IX and the American Welding Society (AWS) D.1.4 Code:

- Containment access door closure weld FW-1;
- Beam (stiffener)-to-containment plate weld FW-1; and
- Splice welds of shield building (SB) rebar joints 2H-03R and 2V-45B.

b. Findings

Inadequate Control of Weld Filler Metal Electrodes

Introduction

A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” were identified by the inspectors for the licensee’s failure to control weld rod oven temperature in accordance with procedure WFMC-1 during a rebar splice weld completed for restoration of the SB access opening.

Description

The inspectors identified that the portable weld rod oven temperature had not been maintained above the minimum required temperature of 250 degrees Fahrenheit (deg F). The inspectors were concerned that if this practice had continued, it would increase the possibility for rebar splice weld failure due to hydrogen-induced cracking.

The coatings of shielded metal arc welding (SMAW) electrodes used for steel (especially low-hydrogen electrodes) readily absorb moisture (i.e., hydroscopic). Water present in the electrode coating, breaks down into hydrogen and oxygen within the welding arc. The hydrogen becomes entrained in the weld metal and as the metal cools, it undergoes a phase transformation from an austenitic to a martensitic structure. From 400 deg F to room temperature, some of the retained austenite changes slowly into martensite (delayed transformation). During this delayed transformation, the monatomic hydrogen has limited solubility and recombines into hydrogen gas causing metal microcracks and fissures. These defects may appear in the weld, at the weld interface, or in the parent metal, depending on how the hydrogen moves or where it becomes trapped and results in delayed hydrogen induced cracks and weld porosity. Because detection of hydrogen-induced cracks is difficult and may not be found until after a weld is placed into service, the controls used to prevent introduction of hydrogen are critical for fabrication of acceptable welds. To prevent introduction of hydrogen, the controls for storage of low-hydrogen electrodes are designed to preclude moisture absorption by the use of hermetically-sealed containers (e.g. shipping package from manufacturer) or by the use of ovens maintained at elevated temperatures to keep the electrode coating dry.

For the low-hydrogen electrodes used to fabricate rebar splices in the restoration of the SB, the licensee’s contractor provided for the control of the low-hydrogen electrodes in procedure WFMC-1 “Bechtel Welding Specification Welding Filler Material Control.” This procedure required the use of portable rod warmers maintained at a minimum temperature of 250 deg F for storage of low-hydrogen electrodes to ensure that the hydroscopic coating of the welding electrode stayed dry and did not absorb moisture from the atmosphere.

During fabrication of a horizontal rebar splice weld 2H-03R, the inspectors observed that the low-hydrogen electrode filler material was protruding several inches above the top of the welder’s portable rod storage oven (e.g. top of oven was open). The inspectors requested the welder verify that the portable oven was at or above the minimum required temperature of 250 deg F. The welder applied a temperature crayon designed to melt at 200 deg F to the oven at the inside surface of the top lid, to the oven inner wall, and at several points on the removable filler metal storage rack. The temperature

crayon did not melt at any of these locations. The welder then removed a single weld electrode and applied the temperature crayon at locations along the weld electrode and was able to get the crayon to melt on the bottom ¼ length of the electrode (portion nearest the bottom of the oven). These measurements demonstrated that the oven temperature had not been maintained above 250 deg F as required by the procedure WFMC-1.

As a corrective action, the licensee removed the welder's certification to weld rebar and documented this issue in CR 2011-05536. The welder who fabricated weld 2H-03R was assigned three additional welds, and absent NRC intervention, these welds would likely have been fabricated with electrodes exposed to ambient temperature conditions for more than 1 hour. A 1-hour time limit outside the warming oven was the maximum allowed by procedure WFMC-1 and AWS D1.4 "Structural Welding Code - Reinforcing Steel" for the E-9018-B3H4R electrode material used on weld 2H-03R to ensure moisture was not absorbed from the atmosphere. To ensure that the horizontal rebar splice weld 2H-03R was not affected by delayed hydrogen cracking, the licensee's vendor examined the weld splice 48 hours after fabrication and did not identify cracks.

Analysis

The inspectors determined that the licensee's failure to control weld rod oven temperature in accordance with procedure WFMC-1 is contrary to 10 CFR 50 Appendix B, Criterion V, and a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Barrier Integrity Cornerstone attribute of Configuration Control and adversely affected the cornerstone objective to provide reasonable assurance that the physical design barriers (e.g. containment) protect the public from radionuclide releases caused by accidents or events. The SB is part of the containment system. Absent NRC identification, rebar welds would have been fabricated with electrodes exposed to ambient temperatures for excessive periods of time creating a condition that results in hydrogen-induced weld cracking. Rebar splice material with cracks returned to service would increase risk for SB failure during design basis events such as wind-driven missile impact or earthquake-induced loads. The inspectors completed a significance determination, in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Containment Barrier. Because the issue was corrected promptly, prior to introduction of weld material with hydrogen induced cracks, the inspectors answered "no" to each of the four Phase 1 screening questions. Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the failure to control the weld rod oven temperature in accordance with procedure WFMC-1 was caused by inadequate licensee oversight of the contracted welder (IMC 0310 - Item H.4(c)). The inspector determined that this was the cause of the finding based upon discussions with licensee and vendor staff.

Enforcement

Appendix B of 10 CFR 50, Criterion V, “Instructions, Procedures, and Drawings,” required in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure WFMC-1, Bechtel Welding Specification Welding Filler Material Control, Revision 0, required; in Paragraph 6.1.2.2 that: “The oven shall be held at a minimum of 250 deg F and a maximum of 350 deg F,” and in Paragraph 7.5 that: “The portable rod warmers shall maintain a minimum temperature of 250 deg F” and in Table 1 for use of “All Low-Hydrogen Electrodes” to “Issue in portable warmers maintained at 250 deg F minimum.”

Contrary to the above, on November 16, 2011, for an activity affecting quality (weld rod oven temperature) the licensee failed to accomplish the activity in accordance with the applicable procedure WFMC-1. Specifically, the licensee failed to maintain a portable rod warmer oven containing low-hydrogen electrode weld material above the minimum required temperature of 250 deg F. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s CAP (CR 2011-05536), it is being treated as a non-cited violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2011005-01)

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The vessel head penetration nozzles and J-groove welds of the inservice head had been affected by primary water stress corrosion cracking (PWSCC) and repaired during the previous outage (reference NRC Inspection Report (IR) 05000346/2010008 – Adams Accession No. ML102930380). For the No. 17 mid-cycle outage, the in-service head was removed to an on-site storage location pending off-site disposal and thus did not require further non-destructive examination. The licensee procured a replacement reactor vessel closure head (RRVCH) with penetration nozzles and J-groove welds fabricated with materials (e.g., Alloy 690) more resistant to PWSCC.

For the RRVCH a bare metal visual pre-service examination and a non-visual pre-service examination was required pursuant to 10 CFR 50.55a(g)(6)(ii)(D) and Code Case N-729-1. The inspectors had previously completed the review of the non-visual preservice examination records for the replacement head as documented in NRC IR 05000346/2011004 (Adams Accession No. ML112991544).

For the pre-service visual examinations of the RRVCH the inspectors observed and reviewed records of the visual examination conducted on the vessel head at penetrations 1, 3, 54, and 61 to determine if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). In particular, the inspectors confirmed that:

- The required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with the licensee procedures;
- The licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and

- That the visual examination procedure required recording indications of potential through-wall leakage and that licensee documented relevant conditions in the corrective action system and implemented appropriate corrective actions.

Prior to the mid-cycle outage, the inspectors observed a welded repair/replacement activity associated with installation of a vent assembly on the upper head penetration of the RRVCH at nozzle No. 21 as documented in NRC IR 05000346/2011004 (Adams Accession No. ML112991544).

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors performed an independent walkdown of portions of the reactor coolant system and attached safety-related systems which had received a boric acid walkdown by the licensee staff to determine whether the licensee's Boric Acid Corrosion Control (BACC) visual examinations emphasized locations where boric acid leaks can cause degradation of safety significant components and to determine if degraded conditions were entered into the CAP.

The inspectors reviewed the following evaluations of reactor coolant system or other safety-related systems with components affected by boric acid to determine if the licensee applied appropriate corrosion rates and properly assessed the effects of corrosion-induced wastage on the component's structural or pressure boundary integrity:

- CR 2010-74892, reactor coolant pump 1-2-2 boric acid;
- CR 2010-79012, high pressure injection (HPI) pump 2P58-2 boric acid; and
- CR 2011-94103, spent fuel pool pump 1-2 seal leak.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR 50, Appendix B, Criterion XVI:

- CR 2010-78548, leak at DH 22A packing;
- CR 2010-73653, leak at DH-11 packing; and
- CR 2010-76667, leak at SF-35 packing.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

Steam Generator tube eddy current (ET) examinations were not required during the No. 17 mid-cycle outage pursuant to TS 3.4.17 "Steam Generator Tube Integrity," and TS 5.5.8 "Steam Generator Program." Therefore, the licensee did not conduct SG tube

examinations and only a portion of the NRC IP could be completed for this review area. Specifically, from October 11, 2011, through November 23, 2011, the inspectors performed an on-site review of documentation related to the SG ISI program to determine if:

- Primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons per day or the detection threshold during the previous operating cycle.

Completion of Section 02.04 of IP 71111.08 is scheduled to be completed during the Spring 2012 refueling outage when the licensee will perform ET of the SG tubes.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

- The licensee had established an appropriate threshold for identifying ISI/SG related problems;
- The licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- The licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Auxiliary building and SB structures; and
- Containment systems.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices;
- Identifying and addressing common cause failures;
- Scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- Characterizing system reliability issues for performance;
- Charging unavailability for performance;
- Trending key parameters for condition monitoring;
- Ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- Verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

The inspectors' maintenance effectiveness reviews constituted two quarterly inspection samples as defined in IP 71111.12-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Work activities during the week ending October 8, 2011, which included a period of yellow shutdown risk during the time that the RCS was drained to reduced inventory conditions. Other activities included the lift and movement of the reactor vessel head from the reactor vessel to the containment storage stand;
- Emergent work associated with cracking identified in the containment SB during the 17M mid-cycle reactor head replacement outage, as documented in CR 2011-03346 and other entries into the licensee's CAP; and
- Emergent repairs associated with damage to decay heat pump 1-1 during the week ending December 24, 2011, as documented in CR 2011-07195.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's

probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Specific documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted three inspection samples as defined in IP 71111.13-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

Decay Heat Pump 1-1 Damaged and Rendered Inoperable By Personnel Climbing on Equipment

Introduction

A self-revealed finding of very low safety significance (Green) was identified when low pressure injection equipment was damaged by operators attempting to access an overhead valve. Specifically, by climbing and standing on sensitive plant equipment, the licensee failed to comply with the standards and expectations for accessing plant equipment contained in procedure NOP-OP-1002, "Conduct of Operations".

Description

On the morning of December 22, 2011, the plant was performing DB-PF-03071, "CCW Check Valve Testing." Performance of this test requires an operator to manipulate CC258, CCW Essential Line 1 to Makeup Pump 1 Isolation Valve. The location of this valve makes it difficult for an operator to gain access. The valve is approximately 12 feet off the floor, amidst overhead interferences, and is located directly above the motor for decay heat removal pump no. 1. A convenient way to access the valve was not readily available when CC258 was required to be closed and opened during the performance of the CCW check valve test. The operator attempting to perform the task determined the most practical method to access the valve was to climb up plant equipment and position himself standing on the top of the decay heat pump motor. The first time the valve was accessed, the operator chose the more open, north, side of the motor. This proved difficult, though the operator was able to make the climb and perform the valve manipulation without event. The operator lowered himself from the motor on the east side, which contained more sensitive equipment, but had more hand and foot holds that made it easier to climb down. For the second time accessing CC258, the operator attempted to climb up the side he had just descended (east), despite the proximity to more sensitive equipment. Upon climbing the motor the second time, the operator's hand slipped causing a fall of approximately 3 feet. The operator landed on his feet, however, during the fall the operator came into contact with the oil temperature probe for the decay heat motor outboard bearing. The temperature element was dislodged and oil began spilling from motor out the open connection.

The loss of oil from the outboard motor bearing rendered low pressure injection pump no. 1 inoperable, causing entry into the action statement for limiting condition for Operation (LCO) 3.5.2.A, which has a 7-day completion time for restoring the system to an operable status. The temperature element was repaired; and the system was

restored in the afternoon of December 22, 2011, after being out-of-service for approximately 13 hours.

The inspectors reviewed the standards and expectations contained in Section 4.17 of NOP-OP-1002, “Conduct of Operations”, covering access to plant equipment. The expectation states that: “Climbing on equipment is the exception and not the rule.” The Conduct of Operations standards also include the following:

- Plant equipment should not be climbed upon to gain access from one location to another. Ladders and/or scaffold are used whenever possible; and
- If no other means are available, plant equipment may be climbed upon provided it does not pose a risk to the safety of personnel or equipment.

Contrary to the standards above, the operator climbed upon plant equipment (decay heat pump motor no. 1) despite the risks involved to personnel safety and equipment safety. An alternate method for accessing CC258 was not addressed in the pre-job brief for the CCW check valve test. A ladder could have been used to provide safer access to the top of the motor or a scaffold addition could have provided better access to the valve itself.

The licensee included this issue in their CAP as CR 2011-07195. An immediate corrective action was taken to repair the damaged temperature element and restore low pressure injection pump no. 1 to operable status. A long-term solution to providing access to the overhead valve is under evaluation in the licensee’s CAP.

<u>Analysis</u>

The inspectors reviewed this finding using the guidance contained in Appendix B, “Issue Screening,” of IMC 0612, “Power Reactor Inspection Reports.” The inspectors determined that the licensee’s failure to comply with the standards and expectations for accessing plant equipment contained in the “Conduct of Operations” procedure was a performance deficiency that was reasonably within the licensee’s ability to foresee and correct and should have been prevented. The inspectors determined that the finding was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Human Performance and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the damage caused when falling from plant equipment rendered low pressure injection train 1 inoperable.

The inspectors evaluated the finding using IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” using the Phase 1 SDP worksheet for the mitigating systems cornerstone. The finding screened as very low safety significance (Green) because the inspectors answered “no” to the screening questions in Table 4a. Specifically, the finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent actual loss of safety function of a single train for greater than its TS allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Work Control component, because the licensee did not plan and coordinate work activities, consistent with nuclear safety. Specifically, the licensee did not appropriately plan for job site

conditions impacting human performance since an appropriate available method for accessing CC258 was not evaluated. (H.3(a))

Enforcement

The inspectors concluded that the licensee did not comply with the standards and expectations for accessing plant equipment contained in procedure NOP-OP-1002, "Conduct of Operations." This finding, however, did not involve a corresponding violation of NRC requirements. Specifically, the inspectors determined that the "Conduct of Operations" procedure is an administrative procedure, and not covered under the quality assurance (QA) requirements set forth in 10 CFR 50, Appendix B. Additionally, the inspectors also determined that the "Conduct of Operations" procedure is not covered under TS 5.4.1(a), which requires the licensee to establish, implement, and maintain applicable written procedures for the safety-related systems and activities recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A. (FIN 05000346/2011005-02)

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- The functionality of the containment SB and operability of the plant's containment system following identification of cracking in the SB concrete, as documented in CR 2011-03346 and other related entries in licensee's CAP;
- The operability of the plant's safety-related station batteries and direct current (DC) electrical distribution systems following identification of loading issues, as documented in CR 2011-01902;
- The functionality and operability of the CCW system following the unexpected drop in CCW surge tank level, as documented in CR 2011-05542; and
- The functionality and operability of the service water (SW) system following issues associated with the balancing of SW train no. 2 safety-related flows after maintenance, as documented in CR 2011-05526.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and USAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

The inspectors' reviews of these operability and functionality evaluations constituted four inspection samples as defined in IP 71111.15-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

Air Voids in Component Cooling Water System Caused By Inadequate Fill and Vent Procedure

Introduction

A self-revealed finding of very low safety significance (Green) and an associated NCV of TS 5.4.1(a) were identified for the licensee's failure to establish, implement, and maintain technically adequate procedures to cover the restoration (i.e., filling and venting) of the CCW system following maintenance activities.

Description

In the early morning hours of November 17, 2011, the plant was in Mode 5, cold shutdown. A heightened shutdown safety awareness condition (i.e., "yellow" risk) was in effect due to the plant having a reduced capacity for decay heat removal until activities to fill and vent the RCS were completed. These RCS filling and venting procedures were in progress.

At approximately 0120 hours, the on-watch control room crew received an unexpected annunciator alarm, 11-3-A, which indicated a low level in the CCW surge tank. The crew entered DB-OP-02011, "Heat Sink Alarm Panel 11 Annunciators," and cut in demineralized water to the CCW system to retard the drop in surge tank level in accordance with the procedure. CCW surge tank level was stabilized, and restored to the normal operating band in short order. The operating crew calculated that approximately 125 gallons of inventory from the CCW surge tank had been lost.

A follow-on investigation by the operating crew revealed no signs of leakage from the system, but that a chemical addition had been made to the CCW system approximately 50 minutes before the receipt of annunciator alarm 11-3-A. From this, the licensee surmised that an air bubble might have been introduced into the CCW system during the chemical addition. The chemical addition piping was a long run of approximately 300 feet that had not been used since maintenance had been conducted on the system and, if voided, could have introduced an air bubble of sufficient size to account for the drop in CCW surge tank level.

On November 18, 2011, the inspectors discussed the issue with the licensee's Superintendent of Nuclear Operations, and voiced a concern regarding how the licensee's procedures for restoration from maintenance activities on the CCW system could have permitted parts of the system to have air entrapped. The possibility that more sections of the CCW system could be voided was also discussed, whereupon the Superintendent of Nuclear Operations stated that the licensee would conduct additional inspections and investigation into the issue.

On November 22, 2011, UT performed by the licensee identified a significant air void, approximately 19 cubic feet, in a CCW pump 3 horizontal suction piping segment. CCW pump 3 is a 'swing' pump that can be lined up to take the place of either the

normal train 1 CCW pump or the normal train 2 CCW pump. Component Cooling Water pump 3 was immediately declared unavailable upon identification of the air void, but because the pump was not lined up to support either train of the CCW system there was no TS implications. Extensive ultrasonic testing was performed on the rest of the CCW system, with no abnormal conditions being noted.

A follow-on investigation by licensee engineering and operations personnel revealed that the air entrapment in the CCW system was most likely due to the extensive maintenance activities on the system during the 17M mid-cycle outage. A complex series of fill and venting evolutions to restore the system had been required, and these evolutions may not have vented all of the air from the system. The licensee had entered this issue into their CAP as CRs 2011-05542 and 2011-05831. Planned corrective actions included additional procedural guidance for CCW fill and venting activities.

<u>Analysis</u>

The inspectors determined that failure of the licensee to establish, implement, and maintain technically adequate procedures to cover the restoration (i.e., filling and venting) of the CCW system following maintenance activities was contrary to the requirements in the licensee's Quality Assurance Program Manual and TS, and as such constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The inspectors reviewed this issue using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," and determined that it was of more than minor safety significance and constituted a finding. The issue was determined to be associated with the Mitigating Systems Cornerstone attribute of procedure quality, and had adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, CCW, a mitigating system, had its reliability adversely impacted by the lack of appropriate fill and venting procedural guidance. In addition, the unexpected 11-3-A annunciator alarm and ensuing alarm response and investigation caused the on-watch operations crew to temporarily suspend the in-progress RCS fill and venting procedures, which extended the heightened shutdown safety awareness condition (i.e., "yellow" risk) by approximately 30 minutes.

The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Because the finding involved reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process - Phase 1 Operational Checklists for Both PWRs and BWRs." Since the finding was associated with an issue that occurred during the time the licensee was conducting RCS fill and venting activities and plant conditions were in transition, the inspectors consulted both Checklist 2, "PWR Cold Shutdown Operation: RCS Closed and Steam Generators Available for Decay Heat Removal (Loops Filled and Inventory in the Pressurizer); Time to Boiling Less Than 2 Hours," and Checklist 3, "PWR Cold Shutdown and Refueling Operation: RCS Open and Refueling Cavity Level Less Than 23 Feet or RCS Closed and No Inventory in the Pressurizer; Time to Boiling Less Than 2 Hours." The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes on either checklist, nor did it meet any

of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee's procedures and guidance for the restoration of the CCW system following outage maintenance activities did not ensure that the system was fully filled and properly vented prior to operation. (H.2(c))

Enforcement

Technical Specification 5.4.1(a) requires the licensee to establish, implement, and maintain applicable written procedures for the safety-related systems and activities recommended in RG 1.33, Revision 2, Appendix A. Section 3(e) of RG 1.33, Revision 2, Appendix A, requires procedures for the proper operation of the CCW system, including filling, venting, and draining operations. Contrary to this requirement, the licensee failed to properly prepare and implement technically adequate written procedures for the filling and venting of the CCW system following mid-cycle outage 17M maintenance, such that significant air voids were left in the system following its restoration and return to service.

Because this finding was of very low safety significance and had been entered into the licensee's CAP as CRs 2011-05542 and 2011-05831, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2011005-03)

1R18 Plant Modifications (71111.18)

.1 Temporary and Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary and permanent plant modifications:

- Engineering change package (ECP) 11-0412, which covered replacement of large portions of service water (SW) piping and removal of emergency core cooling system (ECCS) room cooler check valves [permanent modification];
- ECP 02-0540, which covered replacement of the unit load demand module of the station's integrated control system (ICS) [permanent modification]; and
- ECPs 10-0458 and 10-0459, which covered the opening and restoration of the access openings in the concrete containment SB and steel containment vessel (CV) to facilitate replacement of the integrated reactor head assembly [temporary modification].

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screenings against the design basis, the USAR, and the TS to verify that the modification did not affect the operability or availability of the affected systems. The inspectors observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing

systems. In addition, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

These inspection activities constituted a single temporary modification sample and two permanent plant modification samples as defined in IP 71111.18-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMTs) to verify that procedures and testing activities were adequate to ensure system operability and functional capability:

- Motor testing and pump baseline testing of containment spray train 1 during the week ending October 29, 2011, following motor replacement and preventive maintenance activities;
- Motor testing and baseline testing of no. 1 makeup pump during the week ending October 29, 2011, following motor replacement and preventive maintenance (PM) activities;
- Motor testing and baseline testing of no. 1 decay heat pump during the week ending November 5, 2011, following motor replacement and preventive maintenance activities;
- Motor testing and 18-month response time testing of containment air cooling unit no. 3 during the week ending November 12, 2011, following motor replacement and preventive maintenance activities;
- Post-modification test and 8 hour load test of station battery charger 1N and 1P during the weeks ending October 29 and November 5, 2011, following replacement of the battery chargers;
- Emergency ventilation system train 1 refueling interval SFAS drawdown test during the week ending November 26, 2011, following restoration of the SB and CV openings;
- Testing and tuning of main feedwater regulating valve (FRV) SP6B during the week ending November 5, 2011, following various outage-related maintenance activities;
- Integrated leakage testing of the primary containment during the week ending November 19, 2011, following restoration of the maintenance access opening that facilitated replacement of the reactor vessel integrated closure head assembly;

- Performance testing of auxiliary FW train no. 1 during the week ending December 10, 2011, following various outage-related maintenance activities; and
- Control rod drop timing testing during the week ending December 10, 2011, following replacement of the reactor vessel integrated closure head assembly.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): that the effect of testing on the plant had been adequately addressed; that testing was adequate for the maintenance performed; that acceptance criteria were clear and demonstrated operational readiness; that test instrumentation was appropriate; that the tests were performed as written in accordance with properly reviewed and approved procedures; that equipment was returned to its operational status following testing (i.e., temporary modifications or jumpers required for test performance were properly removed after test completion, etc.); and that test documentation was properly evaluated. The inspectors evaluated the activities against TS, the USAR, 10 CFR 50 requirements, licensee procedures, and various NRC generic communications to verify that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PMTs to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

The inspectors' reviews of these activities constituted ten PMT samples as defined in IP 71111.19-05. In addition, these samples contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

Reactivity Manipulations Performed By Non-Licensed Individual

Introduction

The inspectors identified a SL IV NCV of 10 CFR 54(i). Specifically, on December 4, 2011, during the conduct of control rod insertion timing testing, the inspectors observed a non-licensed member of the licensee's engineering staff operating switches that directly and purposefully caused the insertion of various control rods that were being tested.

Description

On December 4, 2011, the inspectors were observing control rod insertion timing testing as part of a normal baseline inspection sample, and also to fulfill post-installation testing requirements associated with IP 71007, "Reactor Vessel Head Replacement." The sequence of testing involved the withdrawal of each control rod group (one group at a time) from the control room, and then timing the insertion of the control rods upon removal of power from their control rod drive (CRD) mechanisms. This latter action was accomplished locally in the field from electrical penetration room no. 1 where the control rod power supply cabinets were situated.

The inspectors observed the first of several control rod groups to be tested from the control room. Control rod group withdrawal was accomplished by an on-watch licensed

reactor operator who was in constant communication with testing personnel in the electrical penetration room, and no issues were noted by the inspectors. During a brief pause between control rod groups, the inspectors moved to electrical penetration room no. 1 to observe the remaining testing from that location.

During the next control rod group to be tested, the inspectors observed that the actual key switches used to interrupt power to the CRDs and cause control rod insertion were manipulated by a member of the licensee's engineering staff, and not a licensed individual. A licensed Senior Reactor Operator (SRO) was present in the electrical penetration room and providing oversight and test direction. The inspectors immediately questioned the SRO concerning the appropriateness of having a non-licensed individual causing control rod insertion and directly manipulating core reactivity, at which point the testing was suspended and the Shift Manager and Superintendent of Nuclear Operations were contacted and informed of the issue. The licensee immediately dispatched a licensed reactor operator to the electrical penetration room and testing resumed with a licensed reactor operator conducting all further operation of the local key switches.

The licensee entered the issue into their CAP as CR 2011-06318, and initially classified it as a severity level (SL) 5 reactivity management issue (i.e., low level and inconsequential).

<u>Analysis</u>

The inspectors determined that failure of the licensee to assign a licensed operator to manipulate the key switches in the electrical penetration room and directly change core reactivity constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The inspectors reviewed this issue using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports." The issue was determined to be associated with the Mitigating Systems Cornerstone attribute of procedure quality. However, the inspectors subsequently determined that the issue had not adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because of several factors, the inspectors determined that the issue was of minor safety significance and, as such, did not constitute a finding. These factors included:

- All control rod group withdrawal activities were accomplished from the control room by an on-watch licensed reactor operator;
- All activities in the electrical penetration room were performed in accordance with an approved written test procedure, and under the direct supervision of a licensed SRO;
- The operation of the local key switches in the electrical penetration room, albeit by a non-licensed individual, could only cause control rod insertion; there was no withdrawal capability; and
- The individual operating the local key switches in the electrical penetration room was always in continuous communication with the on-watch licensed reactor operator in the control room.

Continuing in Appendix B, “Issue Screening,” of IMC 0612, “Power Reactor Inspection Reports,” the inspectors determined that the issue was subject to the NRC’s traditional enforcement process as an issue that had the potential to impact the agency’s ability to perform its regulatory function. Specifically, the NRC’s Reactor Oversight Process fundamentally assumes that only duly licensed individuals are allowed to manipulate reactor controls and alter core reactivity or make changes to reactor power, and that all licensed individuals perform their licensed duties in accordance with any restrictions associated with their individual licenses.

The inspectors reviewed the violation examples in Section 6.4 of the NRC’s Enforcement Policy, “Licensed Reactor Operators.” However, no similar examples of non-licensed individuals performing licensed duties could be found. Subsequently, the inspectors conferred with NRC Region III management and members of the enforcement staff and determined that, because of the factors noted above, the issue constituted a SL IV violation that resulted in no, or relatively inappreciable, safety consequences. Because this issue was dispositioned through the traditional enforcement process and had no Reactor Oversight Process aspects, there was no cross-cutting aspect associated with the violation.

Enforcement

Controls is defined in 10 CFR 50.2, “Definitions,” as: “When used with respect to nuclear reactors means apparatus and mechanisms, the manipulation of which directly affects the reactivity or power level of the reactor.” Further, 10 CFR 50.54(i) states that: “Except as provided in part 55.13 of this chapter, the licensee may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator as provided in part 55 of this chapter.”

Contrary to this requirement, on December 4, 2011, the licensee permitted a non-licensed member of the engineering staff to manipulate controls (e.g., key switches) in electrical penetration room no. 1 that directly altered core reactivity through the insertion of a group of control rods. Because the licensee entered this issue into the CAP as CR 2011-06318, this SL IV violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2011005-04)

1R20 Outage Activities (71111.20)

.1 Reactor Vessel Head Replacement Outage Activities

a. Inspection Scope

The inspectors reviewed the licensee’s shutdown safety plan and contingency plans for the 17M mid-cycle outage conducted from October 1, 2011, to December 6, 2011, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the shutdown safety plan for key safety functions and compliance with the applicable TS when taking equipment out of service;

- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and the licensee's shutdown safety plan requirements were met, and controls over switchyard activities;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- Controls over activities that could affect reactivity;
- Maintenance of secondary containment as required by TS;
- Licensee fatigue management, as required by 10 CFR 26, Subpart I;
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the containment building to verify that debris had not been left which could block ECCS suction strainers; and
- Licensee identification and resolution of problems related to outage activities.

In addition, the inspectors reviewed the licensee's heavy lift plans and activities in conjunction with the NRC's Operating Experience Smart Sample (OpESS) FY2007-03, Revision 2, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP 71111.20." Documents reviewed during the inspection are listed in the Attachment to this report.

This inspection constituted one non-refueling outage activity sample as defined in IP 71111.20-05. Additionally, these inspection items contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

Inadequate Information on Valve Interlocks Resulted in Inadvertent Operation and Loss of Component Cooling Water Surge Tank Inventory

Introduction

A self-revealed finding of very low safety significance (Green) and an associated NCV of TS 5.4.1(a) were identified for the licensee's failure to establish, implement, and maintain technically adequate procedures to cover the restoration (i.e., motor controller re-energization) of components in the CCW system following maintenance activities.

Description

On Thursday, October 20, 2011, the plant was in a defueled condition. At approximately 1511 hours, the on-watch control room crew received an unexpected annunciator alarm, 11-3-A, which indicated a low level in the CCW surge tank. The crew entered DB-OP-02011, "Heat Sink Alarm Panel 11 Annunciators," and cut in demineralized water

to the CCW system to retard the drop in surge tank level in accordance with the procedure.

At the same time that the above event was occurring, an equipment operator was in the process of restoring electrical loads on 480 Vac Motor Control Center (MCC) E11D. The operator had just closed circuit breaker BE1161 for motor-operated valve (MOV) CC2645, the train 1 auxiliary building return header isolation valve. This valve had been in the shut position to isolate a portion of the CCW system that had been drained for maintenance, and plant operators after reviewing the valve's operating drawings and interlock logic had expected the valve to remain in the shut position following closure of its circuit breaker. As circuit breaker BE1161 was closed, however, CC2645 unexpectedly stroked open. Operations personnel in the control room heard the distinct sounds of collapsing air voids in the CCW piping outside the control room as CC2645 stroked open and annunciator 11-3-A came into alarm.

The on-watch operations crew quickly determined that the unexpected opening of MOV CC2645 was the cause of the low level condition in the CCW system. Because of the low level condition, once CC2645 completed its open stroke it automatically received a command to shut and moved back to the closed position. On an ensuing cycle when the valve was closed or nearly closed, plant operators reopened circuit breaker BE1161 and halted the transient. Component Cooling Water surge tank level was then restored to the normal operating band and stabilized there.

Prior to the event, loads on MCC E11D were being restored at the discretion of the unit supervisor. Drawings being utilized by the plant operators for this activity indicated that CC2645 should only automatically open under a set of very specific conditions. All but one of these conditions were met, and the operators believed that CC2645 would remain in the shut position when circuit breaker BE1161 was closed because an interlock associated with CCW pump no. 1 was not met. Specifically, the operators thought that based on the information on their reference drawings that CC2645 would only open with the circuit breaker for CCW pump no. 1 racked into the "test" position. Since the circuit breaker for CCW pump no. 1 was racked to the "out" position, plant operators had concluded that closing circuit breaker BE1161 would not result in any change in CC2645 valve position. A follow-up investigation by licensee engineering personnel, however, identified that both CCW pump no. 1 being racked into the "test" position and being racked to the "out" position satisfied the CC2645 interlocks and will provide the MOV with a signal to open.

The licensee had entered this issue into their CAP as CR 2011-04078. Corrective action taken or planned by the licensee included a revision to the referenced drawings to include all interlock requirements associated with MOV CC2645, as well as other similar valves.

Analysis

The inspectors determined that failure of the licensee to establish, implement, and maintain technically adequate procedures to cover the restoration (i.e., motor controller re-energization) of the CCW system following maintenance activities was contrary to the requirements in the licensee's Quality Assurance Program Manual and TS, and as such constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The inspectors reviewed this issue using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," and determined that it was of more than minor safety significance and constituted a finding. The issue was determined to be associated with the Mitigating Systems Cornerstone attribute of procedure quality, and had adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, CCW, a mitigating system, had its reliability adversely impacted by the inadequate procedural guidance for motor controller restoration.

The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Because the finding involved reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process - Phase 1 Operational Checklists for Both PWRs and BWRs." Since the finding was associated with an issue that occurred during the time the reactor was in a defueled condition, the inspectors conservatively consulted all four PWR checklists (i.e., Checklists 1 – 4). The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes on any checklist, nor did it meet any of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee's procedures, drawings and guidance for the restoration of the CCW system following outage maintenance activities did not ensure that the system was properly aligned prior to restoration of electrical power to MOV CC2645. (H.2(c))

<u>Enforcement</u>

Technical Specification 5.4.1(a) requires the licensee to establish, implement, and maintain applicable written procedures for the safety-related systems and activities recommended in RG 1.33, Revision 2, Appendix A. Section 3(e) of RG 1.33, Revision 2, Appendix A, requires procedures for the proper operation of the CCW system, including restoration operations following maintenance and other outage activities. Contrary to this requirement, the licensee failed to properly prepare and implement technically adequate written procedures and drawings for the restoration of CCW system components, specifically electrical power to MOV CC2645, following mid-cycle outage 17M maintenance.

Because this finding was of very low safety significance and had been entered into the licensee's CAP as CR 2011-04078, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000346/2011005-05)

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- DB-SC-03121, "Safety Features Actuation System Train 2 Integrated Response Time Test," during the weeks ending October 15, 2011 and November 12, 2011 (routine);
- DB-PF-10310, "Containment Integrated Leakage Rate Test," during the week ending November 19, 2011 (routine);
- DB-SC-03074, "Emergency Diesel Generator 1, ABDC1, and AC103 Appendix R Test," during the week ending October 29, 2011 (routine);
- DB-PF-03010, "Reactor Coolant System Leakage Test," during the week ending December 10, 2011 (RCS leakage);
- DB-PF-03008, "Containment Local Leakage Rate Tests," {Local Leak Rate Test P71C – Core Flood Tank 1-1 Fill and Nitrogen Supply Line and Local Leak Rate Test P49 – Refueling Canal Fill Line}, during the week ending October 8, 2011 (containment isolation valve); and
- DB-SP-03157, "Auxiliary Feedwater Pump 1 Response Time Test," during the week ending December 10, 2011 (inservice testing).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- Did preconditioning occur;
- Were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- Plant equipment calibration was correct, accurate, and properly documented;
- As-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- Measuring and test equipment calibration was current;
- Test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- Test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- Test data and results were accurate, complete, within limits, and valid;
- Test equipment was removed after testing;
- Where applicable for inservice testing (IST) activities, testing was performed in accordance with the applicable version of Section XI, ASME code, and reference values were consistent with the system design basis;

- Where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- Where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- Where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- Prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- Equipment was returned to a position or status required to support the performance of its safety functions; and
- All problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

These inspection activities constituted three routine surveillance testing samples, one RCS leakage testing sample, one containment isolation valve testing sample, and one IST sample as defined in IP 71111.22, Sections -02 and -05. In addition, these samples contributed towards completion of IP 71007, “Reactor Vessel Head Replacement.”

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

The activities in sections 1 through 9 that follow constituted one complete inspection sample as defined in IP 71124.01-05. In addition, these samples contributed towards completion of IP 71007, “Reactor Vessel Head Replacement.”

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the occupational exposure cornerstone for follow-up since the last inspection. The inspectors reviewed the results of radiation protection program audits (e.g., licensee’s QA audits or other independent audits). The inspectors also reviewed reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed and assessed results of the licensee’s audit and operational report reviews to gain insights into overall licensee performance before and during the outage.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors assessed any changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last three to five radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys where appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, containment, fuel handling, and auxiliary building areas to evaluate material conditions, and performed independent radiation measurements to assess conditions of radioactive materials at these areas.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- Reactor head swap work activities in containment;
- ISI of the reactor vessel, core support assembly;
- Plenum and reactor flange maintenance;
- Reactor head disassembly/reassembly work activities; and
- Replacement of SW piping in the auxiliary building.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey programs to determine if hazards were properly identified, including the following:

- Identification of hot particles;
- The presence of alpha emitters;
- The potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials;
- The hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- Severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas during the 17M mid-cycle outage and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors (CAMs) were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or met the requirements of 10 CFR 20.1905(g), “Exemptions To Labeling Requirements.”

The inspectors reviewed the following radiation work permits (RWPs) used to access high-radiation areas and evaluated the specified work control instructions or control barriers.

- Reactor head swap work activities in containment;
- ISI of the reactor vessel, core support assembly;
- Plenum and reactor flange maintenance;
- Reactor head disassembly/reassembly work activities; and
- Refueling activities.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker’s electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP and dose evaluations were conducted as appropriate.

For work activities in transient radiological conditions, the inspectors assessed the licensee’s means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiological control area and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and

evaluated whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the types of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was a procedural guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to assess that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high noise areas as high radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- Reactor head swap work activities in containment;
- ISI of the reactor vessel and core support assembly;
- Plenum and reactor flange maintenance;
- Reactor head disassembly/reassembly work activities; and
- Replacement of SW piping in the auxiliary building.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high efficiency particulate air ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected high radiation areas and very high radiation areas to verify conformance with the occupational PI.

b. Findings

No findings were identified.

.6 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk high radiation areas and very high radiation areas. The inspectors discussed methods employed by the licensee to provide stricter control of very high radiation area access as specified in 10 CFR 20.1602, "Control of Access to Very High Radiation Areas," and Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication before hand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for very high radiation areas and areas with the potential to become very high radiation areas to ensure that an individual was not able to gain unauthorized access to the very high radiation area.

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls.

The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

The activities in Sections 1 through 4 that follow constituted one complete inspection sample as defined in IP 71124.05-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the plant USAR to identify radiation instruments associated with monitoring area radiological conditions including airborne radioactivity, process streams, effluents, materials/articles, and workers. Additionally, the inspectors reviewed the instrumentation system and the associated TS requirements for post-accident monitoring instrumentation including instruments used for remote emergency assessment.

The inspectors reviewed a listing of in-service survey instrumentation including air samplers and small article monitors, along with instruments used to detect and analyze workers' external contamination. Additionally, the inspectors reviewed personnel contamination monitors and portal monitors, including whole-body counters, to detect workers' internal contamination. The inspectors reviewed this instrumentation list to assess whether an adequate number and type of instruments were available to support operations.

The inspectors reviewed licensee and third-party evaluation reports of the radiation monitoring program since the last inspection. These reports were reviewed for insights into the licensee's program and to aid in selecting areas for review ("smart sampling").

The inspectors reviewed procedures that govern instrument source checks and calibrations, focusing on instruments used for monitoring transient high radiological conditions, including instruments used for underwater surveys. The inspectors reviewed the calibration and source check procedures for adequacy and as an aid to smart sampling.

The inspectors reviewed the area radiation monitor (ARM) alarm setpoint values and setpoint bases as provided in the TS and the USAR.

The inspectors reviewed effluent monitor alarm setpoint bases and the calculational methods provided in the Offsite Dose Calculation Manual (ODCM).

b. Findings

No findings were identified.

.2 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors walked down effluent radiation monitoring systems, including at least one liquid and one airborne system. Focus was placed on flow measurement devices and all accessible point-of-discharge liquid and gaseous effluent monitors of the selected systems. The inspectors assessed whether the effluent/process monitor configurations aligned with ODCM descriptions and observed monitors for degradation and out-of-service tags.

The inspectors selected portable survey instruments that were in use or available for issuance and assessed calibration and source check stickers as well as instrument material condition and operability.

The inspectors observed licensee staff performance as the staff demonstrated source checks for various types of portable survey instruments. The inspectors assessed whether high-range instruments were source checked on all appropriate scales.

The inspectors walked down ARMs and CAMs to determine whether they were appropriately positioned relative to the radiation sources or areas they were intended to monitor. Selectively, the inspectors compared monitor response (via local or remote control room indications) with actual area conditions for consistency.

The inspectors selected personnel contamination monitors, portal monitors, and small article monitors and evaluated whether the periodic source checks were performed in accordance with the manufacturer's recommendations and licensee procedures.

b. Findings

No findings were identified.

.3 Calibration and Testing Program (02.03)

a. Process and Effluent Monitors

(1) Inspection Scope

The inspectors selected effluent monitor instruments (such as gaseous and liquid) and evaluated whether channel calibration and functional tests were performed consistent with radiological effluent TS/ODCM. The inspectors assessed whether: (a) the licensee calibrated its monitors with National Institute of Standards and Technology traceable sources; (b) the primary calibrations adequately represented the plant nuclide mix; (c) the sources were verified by the primary calibration when secondary calibration sources were used; and (d) the licensee's channel calibrations encompassed the instrument's alarm set-points.

The inspectors assessed whether the effluent monitor alarm setpoints were established as provided in the ODCM and station procedures.

When changes to effluent monitor setpoints were made, the inspectors evaluated the bases for the changes to ensure that an adequate justification existed.

(2) Findings

No findings were identified.

b. Laboratory Instrumentation

(1) Inspection Scope

The inspectors assessed laboratory analytical instruments used for radiological analyses to determine whether daily performance checks and calibration data indicated that the frequency of the calibrations was adequate and there were no indications of degraded instrument performance.

The inspectors assessed whether appropriate corrective actions were implemented in response to indications of degraded instrument performance.

(2) Findings

No findings were identified.

c. Whole Body Counter

(1) Inspection Scope

The inspectors reviewed the methods and sources used to perform whole body count functional checks before daily use of the instrument and assessed whether check sources were appropriate and aligned with the plant's nuclide mix.

The inspectors reviewed whole body count calibration records and evaluated whether calibration sources were representative of the plant source term and whether the appropriate calibration phantoms were used. The inspectors assessed the calibration data for anomalous results or other indications of instrument performance problems.

(2) Findings

No findings were identified.

d. Post-Accident Monitoring Instrumentation

(1) Inspection Scope

The inspectors selected containment high-range monitors and reviewed the calibration documentation since the last inspection.

The inspectors reviewed the electronic calibration data and assessed whether calibration acceptance criteria were reasonable, accounted for the large measuring range, and reflective of the intended purpose of the instruments.

The inspectors reviewed the licensee's stack effluent process monitors that were relied on by the licensee in its emergency operating procedures as a basis for triggering emergency action levels and emergency classifications in order to make protective action recommendations during an accident. The inspectors evaluated the calibration and availability of these instruments.

The inspectors reviewed the licensee's capability to collect high-range, post-accident iodine effluent samples.

As available, the inspectors observed electronic and radiation calibration of these instruments to assess conformity with the licensee's calibration and test protocols.

(2) Findings

No findings were identified.

e. Portal Monitors, Personnel Contamination Monitors, and Small Article Monitors

(1) Inspection Scope

During a review of these instruments used on site, the inspectors assessed whether the alarm setpoint values were reasonable under the circumstances to ensure that licensed material was not released from the site.

The inspectors reviewed the calibration documentation for each instrument selected and discussed the calibration methods with the licensee to determine consistency with the manufacturer's recommendations.

(2) Findings

No findings were identified.

f. Portable Survey Instruments, Area Radiation Monitors, Electronic Dosimetry, and Air Samplers/Continuous Air Monitors

(1) Inspection Scope

The inspectors reviewed calibration documentation for at least one of each type of instrument. In reviewing these portable survey instruments and ARMs, the inspectors reviewed detector measurement geometry and calibration methods and had the licensee demonstrate use of its instrument calibrator as applicable. The inspectors conducted comparison of instrument readings versus an NRC survey instrument if problems were suspected.

As available, the inspectors reviewed the data for portable survey instruments that did not meet acceptance criteria during calibration in order to assess whether the licensee took appropriate corrective actions with instruments that were found significantly out of calibration greater than 50 percent. The inspectors assessed whether the licensee evaluated the out of tolerance instruments for possible consequences when used during radiation surveys.

(2) Findings

No findings were identified.

g. Instrument Calibrator

(1) Inspection Scope

As applicable, the inspectors reviewed the current output values for the licensee's portable survey and ARM instrument calibrator units. The inspectors assessed whether the licensee periodically measures calibrator output over the range of the instruments used through measurements by ion chamber/electrometer.

The inspectors assessed whether the measuring devices had been calibrated by a facility using National Institute of Standards and Technology traceable sources and whether corrective factors for these measuring devices were properly applied by the licensee in its output verification.

(2) Findings

No findings were identified.

h. Calibration and Check Sources

(1) Inspection Scope

The inspectors reviewed the licensee's 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," source term to assess whether calibration sources used were representative of the types and energies of radiation encountered in the plant.

(2) Findings

No findings were identified.

.4 Problem Identification and Resolution (02.04)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring instrumentation were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring instrumentation.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Occupational Radiation Safety, Public Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity performance indicator for the period from October 2010 through September 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, CRs, and NRC Integrated Inspection Reports for the period from October 2010 through September 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's condition report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one reactor coolant system specific activity sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage performance indicator for the period from October 2010 through September 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, condition reports and NRC Integrated IRs for the period from October 2010 through September 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's condition report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one reactor coolant system leakage sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 <u>Semi-Annual Trend Review</u>

a. <u>Inspection Scope</u>

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6 month period of July 1, 2011, through December 31, 2011, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, QA audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted a single semi-annual trend inspection sample as defined in IP 71152-05.

b. <u>Observations</u>

The inspectors identified a potential adverse trend related to the technical quality of the licensee's infrequently performed procedures, specifically those procedures that are potentially only utilized during a unit refuel or other outage:

- On October 20, 2011, the on-watch control room crew received an unexpected annunciator alarm, 11-3-A, which indicated a low level in the CCW surge tank. The crew entered DB-OP-02011, "Heat Sink Alarm Panel 11 Annunciators," and cut in demineralized water to the CCW system to retard the drop in surge tank level in accordance with the procedure. The lowering level in the CCW surge tank was traced to the inadvertent stroking of MOV CC2645, the train 1 auxiliary building return header isolation valve, which had unexpectedly stroked open when plant operators restored 480 Vac power to the MOV as part of system restoration following outage maintenance. Operations personnel had been misled by instructional notes on approved plant drawings that indicated that the valve would not automatically stroke open upon restoration of power. A finding associated with this issue is discussed in detail in Section 1R20.1 of this report;

- On November 17, 2011, the on-watch control room crew again received an unexpected annunciator alarm, 11-3-A, which indicated a low level in the CCW surge tank. Once again, the crew entered DB-OP-02011, "Heat Sink Alarm Panel 11 Annunciators," and cut in demineralized water to the CCW system to retard the drop in surge tank level in accordance with the procedure. Following this event, the lowering level in the CCW surge tank was traced to air intrusion into the CCW system. A complex series of fill and venting evolutions to restore the system had been required, and these evolutions had not vented all of the air

from the system. A finding associated with this issue is discussed in detail in Section 1R15.1 of this report;

- On November 16, 2011, an Alert was declared by the licensee due to a fire and "explosion" with a "flash of flame" coming from safety-related MCC E11C. The fire was caused by an electrical short within one of the MCC's circuit breakers that had resulted from water intrusion. A demineralized water supply valve, PW55, located above MCC E11C had been overpressurized and leaked water onto the MCC. The overpressurization of PW55 resulted from an improper sequence of procedure steps for the switching of makeup water to the station's auxiliary boiler that relied upon a check valve to protect lower pressure rated piping and components. A finding associated with this issue is discussed in detail in Section 4OA3.2 of this report;

- On November 21, 2011, while conducting RCS fill and venting activities the licensee overpressurized the low-range suction pressure gauges on decay heat pump 1 and decay heat pump 2. The sequence of steps in procedure DB-OP-06904, "Shutdown Operations," was identified as the issue. The licensee documented this issue in their CAP as CRs 2011-05781 and 2011-05782;

- On November 19, 2011, the licensee attempted to obtain a chemistry sample to verify pressurizer dissolved oxygen concentrations during pressurizer heatup in accordance with procedure DB-CH-06002, "Sampling System Nuclear Areas." The sample was unable to be obtained due to limitations with the procedure as written. The licensee documented this issue in their CAP as CR 2011-05726; and

- On November 29, 2011, the licensee identified an adverse condition associated with procedure DB-PF-03811, "Miscellaneous Valves Test." The procedure as written would have overpressurized a section of piping in the decay heat system had it been performed as scheduled. Fortunately, the on-watch Operations crew identified the vulnerability before the procedure was performed and had it rescheduled for a time when plant conditions would adequately support it. The licensee entered the issue into their CAP as CR 2011-06011.

In each of these cases, the issue was of very low or minor safety significance. However, taken collectively they represent a potential adverse trend that may require a mitigation strategy.

c. Findings

No findings were identified.

.4 Annual Sample: Review of Licensee Extent-of-Condition for Shield Building Concrete Cracking

a. Inspection Scope

On October 10, 2011, a laminar crack was found in the flute shoulder area of the opening being cut through the SB concrete cylindrical wall for transfer of the RRVCH head. The crack was found on the vertical side of the opening (left side, looking from the

outside), generally along the main reinforcing steel of the cylinder, and extending to across the top (approximately 6 feet) and across the bottom (approximately 4 feet) of the opening. After the licensee performed some minor manual chipping along the edges, the crack indication along the left and bottom edges essentially disappeared. Based on the observation, the licensee considered the crack to have been a circumferential laminar tear and not a radial 'through-thickness' direction crack. The licensee initiated CR 2011-03346 to identify this issue and informed the NRC via the Resident Inspectors' Office on site.

Based upon inspection of this crack at the SB opening, the licensee determined that the extent of the cracking warranted further examination and investigation. A contractor was contacted to perform impulse response (IR) testing. The IR testing method measured the structure's frequency at a specific location and plotted that frequency with adjacent readings to obtain any change in building frequency. Changes in frequency within a short span were possible subsurface indications of concrete cracking. To confirm the IR readings, the licensee performed core boring of the concrete in the indicated areas (where cracking was suspected) and in the adjacent areas (where no cracking was suspected). The IR readings were performed on a representative sample of all readily accessible areas of the SB, with the progression of IR testing based upon the indications of possible cracking that were obtained. From this information, the licensee concluded that the laminar cracking initially identified adjacent to the RRVCH opening was not restricted to that area, but was a much more generic issue for the SB as a whole. The licensee entered this extent-of-condition issue for the SB cracking into their CAP as CR 2011-03996 on October 19, 2011, and informed the NRC via the Resident Inspectors' Office on site.

On October 26, 2011, during investigation actions associated with CR 2011-03346, the licensee identified additional areas of concern via IR testing in semicircular zones above the main steam line penetrations through the SB. This condition appeared to be different from the condition documented in CR 2011-03346, which had been primarily concerned with cracking at the SB opening and similar areas around the building's circumference. These new areas of concern were not similar to those previously identified, and appeared to be associated with the main steam line penetrations. The licensee entered this extent-of-condition issue for the SB cracking into their CAP as CR 2011-04402 and informed the NRC via the Resident Inspectors' Office on site.

On October 31, 2011, the licensee identified additional indications of concrete cracking during IR testing towards the top of the SB wall, approximately between the 780 ft and 800 ft elevations. This area of indications was yet another one different from the laminar cracking initially identified adjacent to the RRVCH opening. The licensee entered this extent-of-condition issue for the SB cracking into their CAP as CR 2011-04648, informed the NRC via the Resident Inspectors' Office on site, and continued to investigate further to determine if any additional adverse conditions existed.

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve these challenges. The inspectors also reviewed the associated CRs and investigations for the issue to verify that the licensee's identification of the problems were complete, accurate, and timely, and that the consideration of the extent-of-condition reviews, generic implications, common causes, and previous occurrences, if any, were adequate. Throughout the entire process, the NRC Resident Inspectors' Office was augmented by structural engineering experts from the NRC

Region III Office in Lisle, Illinois, as well as structural engineering and concrete construction experts from the Office of Nuclear Reactor Regulation located at NRC Headquarters in Washington, D.C.

Inspector follow-up activities related to the laminar concrete cracking and the long-term impact on the SB are on-going. Upon completion, the inspection will be documented in a separate report, IR 05000346/2012007, along with the results of the NRC's technical assessment of the licensee's evaluation of the SB's capability to perform its designated safety functions.

The documents listed in the Attachment were reviewed to accomplish the objectives of the IP. This review constituted one annual inspection sample as defined in IP 71152-05. In addition, this sample contributed towards completion of IP 71007, "Reactor Vessel Head Replacement."

b. Findings

No findings were identified.

.5 Selected Issue Follow-Up Inspection Associated with Condition Report 07-26185 "Degradation Found on Rip-Rap sides of the Forebay and Intake Canal"

a. Inspection Scope

The inspectors reviewed the corrective actions associated with excessive settlement in a section of the safety-related Northern Wall of the Intake Canal Forebay that was identified by the licensee and documented in CR 07-26185 "Degradation Found on Rip-Rap sides of the Forebay and Intake Canal." This issue was selected for an in-depth review based on the Ultimate Heat Sink inspection (Section 1R.07) and discussions with the Division of License Renewal in the Office of Nuclear Regulation. The inspectors reviewed the troubleshooting activities and subsequent CRs to verify that the licensee was appropriately addressing the adverse condition in their corrective action program.

b. Findings and Observations

During a routine inspection of the Intake Canal in 2007, the licensee identified unexpected settlement on the North side of the embankment in the safety-related portion of the Forebay for a length of approximately 200 feet. This settlement reduced the slope of the embankment and the concern was captured in CR 07-26185. The licensee contracted an external organization to perform a stability study to ensure the operability of the embankment. In 2009, the licensee received the final report and CR 09-54330 "Slope Stability Study for the FOREBAY North Wall Found Low Strength Clay Till," was created to evaluate the conclusions of the contractor's report, Bowser- Morner Report No. 144188-0209-1575. The report documented the soil profile of the core bores taken above the affected areas were very similar to the soil profile described in the original plant design documents. The licensee concluded the condition was insignificant and did not affect operability of the canal walls; however, the licensee initiated actions to restore the canal wall. However, in 2010, the licensee rejected the initial repair plan and in March 2011, concluded additional data and analysis were necessary to understand the cause of the condition.

During the Ultimate Heat Sink inspections, the inspectors walked the length of the canal walls with the system engineer and noted that the wall had degraded further. At the NRC's prompting, the licensee initiated CR 11-97166, "Degradation of the Intake Canal North Wall in the Q/NQ Portion of the Canal." The licensee concluded the canal remained operable based on EQE Calculation 250785-C-001, "Slope Stability," dated March 31, 1999, which evaluated erosion of the earthen wall embankment in the nonsafety-related portion of the canal. However, the inspectors questioned the applicability of the EQE Calculation and subsequent licensee conclusion of current functionality because the EQE addressed a specific failure mechanism, which was not present in the safety-related section of the intake structure. The inspectors also questioned the term "stable" being used to describe what the inspectors observed as active degradation of the canal wall.

During this same time period, the licensee performed a surveillance to measure the length and width of the intake canal. The licensee concluded portions of the canal were narrower than expected; therefore, the Intake Canal did not meet the licensed design requirements due to volume reduction of approximately three percent (3 percent). Additionally, the slopes, canal toe-to-toe lengths and wall heights were not consistent with the original design requirements and documentation. The licensee initiated CR 11-00422 "Intake Canal Dike Does Not Meet Design Configuration Requirements," and a prompt operability determination. The licensee re-calculated the available volume and surface area and determined that margin was available such that the canal remained operable. However, the licensee noted UFSAR Section 2.5.1.10.2, "Foundations for Seismic Class I Structures; Seismic Class I Intake Forebay Canal Dikes," stated the intake canal was designed to have a 2.5 factor of safety against failure during the maximum possible earthquake. The current condition resulted in a factor of safety of 2.44. The licensee determined in the prompt operability determination that while this condition did not meet the UFSAR, there was reasonable assurance of operability.

Although the licensee was aware of the above issues, the inspectors were concerned that in the prompt operability determinations, the licensee narrowly assessed the current, as-found condition and did not consider whether the mechanism causing the degradation could result in a problem beyond additional sediment in the canal. As part of their corrective actions, core bores of the affected area were obtained and the November 2011 preliminary assessment found the soil profile was very similar to that described in the original plant design documents. The inspectors were also concerned with the timeliness of repairs described in the CRs because the initial plans were replaced with continued monitoring. The condition and the licensee's plans to repair the wall became the subject of several Requests for Additional Information (RAIs) associated with the License Renewal Application. As documented in a letter dated October 31, 2011, (ML11306A066), the licensee outlined a schedule to repair the canal wall while continuing to monitor for further degradation.

No findings were identified.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152 05.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report 05000346/2011-001-00: Pressurizer Code Safety Valve Setpoint Test Failures

On February 28, 2010, Davis-Besse commenced refueling outage sixteen. Per the outage plan, the site's pressurizer safety valves were removed and sent to an offsite vendor for testing and refurbishment. This testing was performed on August 16, 2010. In December 2010, the licensee received information from the testing vendor that the two pressurizer safety valves had as-found lift setpoints (2531 psig and 2535 psig respectively) that were slightly above the limits specified in TS 3.4.10 (2525 psig). The licensee attributed the as-found values to setpoint drift during operation. A past operability evaluation was completed by the licensee on January 12, 2011, and concluded that the pressurizer safety valves had been inoperable while they were installed in the plant during the previous reactor operating cycle.

The inspectors' review of this event determined that the safety significance of the issue was minimal. While both valves had as-found setpoints that exceeded the TS allowed value, the highest out-of-tolerance setpoint was only 10 psig higher than the required value, and the discrepancy would not have adversely impacted either valve's ability to have fulfilled its safety function had either been called upon to do so during the previous period of reactor operation. Consequently, the inspectors determined that this failure to comply with TS 3.4.10 was a violation of minor safety significance that was not subject to formal enforcement action in accordance with Section 2.3 of the NRC Enforcement Policy.

The licensee had entered these failures into their CAP as CR 2010-87048. Documents reviewed as part of this inspection are listed in the Attachment. This Licensee Event Report (LER) is closed.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153-05.

.2 Event Notification 47443: ALERT Due to Fire in Electrical Bus Affecting Safety-Related Equipment

a. Inspection Scope

In the early morning hours of November 16, 2011, the inspectors responded to the site following the report of an electrical explosion and fire, and the licensee's declaration of an Alert per the site's Emergency Plan. In response to the event, the inspectors observed and reviewed the licensee's response to the event, plant parameters and status, including but not limited to:

- Mitigating systems and fission product barriers performance and integrity;
- The realignment of the plant's affected electrical equipment;
- All emergency notifications made to state and local government agencies as required by 10 CFR 50.72; and
- Emergency plan termination and exit.

The inspectors remained on station in the site's control room providing independent assessment of the event until after the licensee had completed determinations that the

Alert could be terminated. Documents reviewed in this inspection are listed in the Attachment.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153-05.

b. Findings

Inadequate Procedure Resulted in Water Intrusion Into Safety-Related Motor Control Center

Introduction

A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to establish, implement, and maintain technically adequate procedures to permit the proper switching of FW sources for the station's auxiliary boiler, such that when the switching of FW sources from demineralized water to the station's normal condensate system took place per approved procedures there were detrimental results. Specifically, the approved procedures for this activity relied upon a check valve to keep the demineralized water header from being exposed to greater pressure than its design. When the check valve failed to function as designed, a failure of demineralized water system components and the inadvertent deluge and failure of safety-related electrical equipment resulted.

Description

At 0200 hours, the licensee switched makeup water to the auxiliary boiler from the demineralized water system, operating at approximately 95 psig, to the plant's normal condensate system, operating at approximately 300 psig. Shortly thereafter, at approximately 0205 hours, control room operators were notified of water spraying from the overhead in the auxiliary building corridor between mechanical penetration rooms 3 and 4. The operations Shift Engineer, who was dispatched to the scene, reported that the water was coming down on safety-related MCC E11C.

At about 0214 hours, the Shift Engineer witnessed an "explosion" with a "flash of flame" coming from MCC E11C. The control room was notified and operations personnel entered the site's procedures for a fire and dispatched the fire brigade. Electrical power was removed to MCC E11C by opening the feeder breaker to the entire E11 MCC string (i.e., E11A, E11B, E11C, E11D, and E11E). Power to numerous train 1 safety-related MOVs was lost as a result. However, because the plant was in cold shutdown/Mode 5 at the time of the event, these MOVs were either not required to be operable under present plant conditions and/or already in their safety-related positions. At 0222 hours, the on-watch Shift Manager declared an Alert per the site's Emergency Plan in accordance with Emergency Action Level HA4, "Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown."

The fire was reported out by the fire brigade at approximately 0233 hours. No extinguishing agents were required; the removal of electrical power resulted in the fire burning itself out. A subsequent investigation of the condition of MCC E11C revealed that the source of the fire was the 480 Vac breaker BE1144, "HA5261A Control Room Emergency Ventilation Fan 1-1 Inlet Valve," and that certain breaker subcomponents had shorted when they became wetted by the water spray cascading down through the

MCC from the overhead. Further investigation revealed that the source of the water was a small diaphragm valve located above MCC E11C, PW55, which served to supply demineralized water to a nearby maintenance shop. At approximately 0443 hours, the site exited from the Alert and fire response procedures.

Following the event, the licensee conducted an investigation into the cause. Station engineering personnel quickly concluded that the procedure being used to switch makeup water to the auxiliary boiler from the demineralized water system, operating at approximately 95 psig, to the plant's normal condensate system, operating at approximately 300 psig, contained a sequence of steps that relied upon a check valve to keep portions of the demineralized water system from being exposed to the much higher condensate system pressure. When the check valve failed to completely close, the excessive pressure, albeit not high enough to damage piping and other "hard" components within the demineralized water system due to that system's installed relief valve protection, was high enough to cause catastrophic failure to "soft" components, such as the soft diaphragm inside PW55. More egregious, however, was that there were no fewer than three previous occurrences (March 7, 2006, December 31, 2007, and August 28, 2008) where the licensee had identified water spraying from above MCC E11C under similar circumstances, but failed to pursue the matter sufficiently to identify the real cause and enact proper corrective actions. The licensee had entered this issue into their CAP as CRs 2006-00624, 2007-32157, 2008-45463; 2011-05456, 2011-05457, 2011-05465, 2011-05466, and 2011-05523. Corrective actions taken by the licensee included, but were not limited to, changes to the station auxiliary boiler operating procedure and repair of the affected electrical components.

Analysis

The inspectors determined that failure of the licensee to establish, implement, and maintain technically adequate procedures to permit the proper switching of FW sources for the station's auxiliary boiler was contrary to the requirements in the licensee's administrative procedure governing the content of balance-of-plant system procedures, NG-QS-00121, "Davis-Besse Procedure Writer's Guide," and as such constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The inspectors reviewed this issue using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," and determined that it was of more than minor safety significance and constituted a finding. The issue was determined to be associated with the Initiating Events cornerstone attribute of procedure quality, and had adversely affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, electrical power to an entire string of safety-related 480 Vac MCCs (i.e., E11A, E11B, E11C, E11D, and E11E) was forced to be interrupted when a deficient procedure for the operation of the station's auxiliary heating boiler caused a significant amount of water to be deluged onto MCC E11C, resulting in an electrical short and fire within the MCC.

The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Because the finding involved reactor shutdown operations and conditions, the inspectors transitioned to IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process –

Phase 1 Operational Checklists for Both PWRs and BWRs." Since the finding was associated with an issue that occurred during the time the licensee was in a cold shutdown (Mode 5) condition, the inspectors consulted Checklist 3, "PWR Cold Shutdown and Refueling Operation: RCS Open and Refueling Cavity Level Less Than 23 Feet or RCS Closed and No Inventory in the Pressurizer; Time to Boiling Less Than 2 Hours." The inspectors determined that the finding did not adversely impact any shutdown defense-in-depth or mitigation attributes, nor did it meet any of the checklist specific requirements for a Phase 2 or Phase 3 SDP analysis. Consequently, the finding was determined to be of very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, CAP component, because the licensee did not take appropriate corrective actions to address the safety issue in a timely manner, commensurate with the safety significance and complexity. Specifically, the licensee had multiple previous opportunities to have appropriately diagnosed and corrected the issue, but failed to do so. (P.1(d))

Enforcement

The inspectors concluded that the licensee did not comply with the standards and expectations for establishing, implementing, and maintaining technically adequate procedures to permit the proper switching of FW sources for the station's auxiliary boiler, as required in Attachment F7 of NG-QS-00121, "Davis-Besse Procedure Writer's Guide." This finding, however, did not involve a corresponding violation of NRC requirements. Specifically, the inspectors determined that the "Davis-Besse Procedure Writer's Guide" is an administrative procedure, and not covered under the QA requirements set forth in 10 CFR 50, Appendix B. Additionally, the inspectors also determined that the "Davis-Besse Procedure Writer's Guide" is not covered under TS 5.4.1(a), which requires the licensee to establish, implement, and maintain applicable written procedures for the safety-related systems and activities recommended in RG 1.33, Revision 2, Appendix A. (FIN 05000346/2011005-06)

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000346/2011-004-01: Plant Transient During High Pressure Injection Flow Instrument String Checks

On September 15, 2011, instrumentation and controls (I&C) technicians replaced the HPI 3A and 3B flow instrument signal monitors with refurbished modules. Upon insertion of the module into the cabinet, the control room received an unexpected alarm for ICS Input Mismatch. The alarm immediately cleared and was attributed to a slight disruption in voltage when the modules were inserted. A decision was made to continue replacement activities. On September 16, 2011, I&C technicians commenced PMT of the signal monitors. During the string check of the HPI flow instrument alarms, annunciator alarm 14-4-E, "ICS Input Mismatch," was received. The alarm initially cleared, then returned. Coincident with ICS Input Mismatch alarm, the plant's ICS began reducing reactor power without any operator input. On-watch plant operators entered procedure DB-OP-02526, "Primary to Secondary Plant Upset," and went through actions of placing ICS stations in manual control. The I&C technicians performing the HPI flow instrument signal monitor refurbishment were directed to stop their activities. Reactor power initially dropped to approximately 95 percent before

operators stabilized the plant, and then returned reactor power to approximately 100 percent using manual controls.

The refurbished HPI flow instrument signal monitor modules were removed from the system and taken to the I&C shop for inspection and testing, while the original signal monitor modules were reinstalled. Inspection and testing of the refurbished modules in the I&C shop did not reveal any issues. The modules were sent to the licensee's offsite testing laboratory for further analysis.

The inspectors reviewed the licensee's analysis which identified that when the K1 "high flow" positive relay coil was energized on the refurbished signal monitor module, Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) affected the low input ICS converter module located in an adjacent slot in the cabinet. The licensee's laboratory identified that some of the signal monitor modules, including the one used that caused the plant transient, did not have ferrite suppression beads on the leads of two capacitors in the circuitry. Ferrite suppression beads are passive electrical components used to suppress high frequency noise and prevent oscillations from occurring. The licensee's laboratory confirmed that signal monitor circuit boards without ferrite beads resulted in oscillations larger in duration and amplitude than circuit boards that did contain ferrite suppression beads. These oscillations were the underlying reason why EMI/RFI was generated from the K1 "high flow" relay coil, causing the ICS circuitry to respond.

The installation of ferrite suppression beads on signal monitor circuit boards was an enhancement that the circuit card manufacturer had implemented sometime in the 1980's. Circuit boards manufactured in the 1970's did not contain ferrite beads. The licensee's supply of signal monitor modules contains a mix of boards with and without the ferrite suppression beads installed. The licensee indicated that they did not have any prior knowledge of this design enhancement and discovered it during the investigation of the event. A review of operating experience did not reveal any similar design issues associated with the signal monitor modules at Davis-Besse or any other nuclear plant facility. Therefore, the inspectors determined that the issue was a latent problem with the refurbished circuit board and was not within the licensee's ability to foresee and correct. The licensee has initiated corrective actions to inspect all currently installed signal monitor modules of the same module and will replace boards that do not contain ferrite suppression beads. Also, an order was created to inspect all spare signal monitor modules onsite to identify any other boards that lack ferrite suppression beads. All further circuit board refurbishments at the laboratory will contain a requirement to ensure ferrite suppression beads are installed.

The inspectors did not identify a performance deficiency or violation of NRC requirements. Based on the inspectors' review of the licensee's analysis of the event, this unresolved item is closed.

.2 Reactor Vessel Head Replacement (IP 71007) – Containment Access Restoration

a. Inspection Scope

The Davis-Besse containment lacked an access opening of sufficient size to permit removal of the old vessel head and reinstallation of the RRVCH. Therefore, the licensee cut a temporary access opening in the SB and CV of sufficient size to support the head replacement. To restore the temporary construction opening in the CV, the licensee

reused and reinstalled (by SMAW) the original plate section cut from the CV. To restore the temporary construction opening in the SB, the licensee installed new reinforcing steel (i.e., rebar) to replace the original steel reinforcement and poured new concrete fabricated at an on-site batch plant.

The inspectors reviewed the licensee activities associated with the restoration of the CV and SB access openings. Specifically, the inspectors observed activities and reviewed records as discussed below:

- Inspectors observed the cutting of the CV opening using a track-mounted welding torch to determine if the cutting activity followed the WO;
- Inspectors observed installation of the replaced CV plate to determine if the gap tolerances had been maintained in accordance with the WO and to determine if site procedures were adequate to control plate distortion;
- Inspectors observed full penetration butt welds fabricated during reinstallation of the 1.5 inch thick CV access plate to determine if the welding process followed the qualified welding procedures and to determine if weld filler materials were traceable to certified material test reports;
- Inspectors reviewed the welding procedures and welder qualification records for containment closure welding activities to determine if the welding was qualified in accordance with the ASME Code Section IX;
- Inspectors reviewed samples of the radiographic (RT) records and magnetic particle (MT) exam records of the CV welds to determine if weld acceptance criteria met the CC requirements (ASME Code 1968 Edition, 1969 Summer Addenda of Section III);
- Inspectors observed installation of mechanical rebar splices (reattachment by crimping of the steel reinforcement (rebar)) in the reinforcing steel used to restore the SB opening to determine if the licensee process conformed to the qualified procedure and design requirements;
- Inspectors observed installation of welded rebar splices in the reinforcing steel used to restore the SB to determine if the welding process followed the qualified welding procedures and that weld filler materials were traceable to certified material test reports and that welders were properly qualified;
- Inspectors reviewed the results of concrete field tests (e.g., slump and air content) during installation to determine if the concrete had the expected properties specified for the mix design;
- Inspectors observed the onsite and off-site storage and curing conditions for concrete test cylinders to determine if they met the American Society for Testing and Materials (ASTM) C31 “Making and Curing Concrete Test Specimens in the Field,” and ASTM C192 “Making and Curing Concrete Test Specimens in the Laboratory,” prior to acceptance testing;
- Inspectors reviewed the licensee’s vendor records for the source materials (e.g., aggregate, cement, water, and admixtures) for concrete batches used in restoration of the SB to determine if it these materials conformed to the design specifications;
- Inspectors observed concrete cylinder compressive tests to determine if testing was conducted in accordance with ASTM C39 “Compressive Strength of Cylindrical Concrete Specimens,” and to determine if the test results demonstrated that the concrete used for restoration of the SB opening had

adequate shear strength to meet the USAR Section 3.8.2.3.7 minimum design compressive strength (e.g., in excess of 4000 psi); and

- The records reviewed by the inspectors are identified in the Attachment to this report.

b. Findings

No findings were identified.

.3 (Closed) Unresolved Item 05000346/2011004-05: Code Surface Examination Requirements Not Applied to Closure Head Stud Holes

a. Inspection Scope

During the review of the fabrication records for the RRVCH, the inspectors identified a URI associated with the licensee's decision to not perform surface examination of the accessible surfaces of the RRVCH stud holes based upon an interpretation of the ASME Code Section III requirements. On October 6, 2011, the Agency completed a review of the licensee's interpretation of the Code, and determined that it was not correct (reference Task Interface Agreement (TIA) No. 2011-15 - ADAMS Accession No. ML11279A218). Based upon review of this issue as discussed below, Unresolved Item (URI) 05000346/2011004-05 is closed.

b. Findings

Incomplete Surface Examination of the Replacement Reactor Vessel Closure Head

Introduction

A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," were identified by the inspectors for the licensee's failure to perform an adequate review of fabrication records to ensure material procured from a contractor (RRVCH) met the CC. Specifically, the accessible surfaces of the 60 closure head flange stud holes were not subjected to PT or MT examinations as required by the CC.

Description

The inspectors identified that the licensee had not performed PT or MT examinations of accessible surfaces for the 60 closure head flange stud holes as required by the CC. The inspectors were concerned that failure to perform these examinations could have allowed rejectable indications to be placed inservice.

On July 22, 2011, during review of RRVCH fabrication records, the inspectors identified that the licensee had not completed the surface examinations required by the CC (1989 Edition of the ASME Code Section III). Specifically, the accessible surfaces of the RRVCH flange stud holes had not been examined using MT or PT methods as required by the Articles NB-2541(a) and NB-4121.3 of Section III of the ASME Code. The inspectors were concerned that without surface examination, rejectable flaws could be placed in service. Additionally, inservice examination of stud hole surfaces is not required by Section XI of the ASME Code, so rejectable fabrication defects would not be identified once the RRVCH was placed inservice. In response to the inspectors'

concern, the licensee determined that the accessible interior surfaces of the RRVCH stud holes did not require surface examination. The licensee's position was based on the ASME Code Interpretation III-1-77-162, which stated in part that drilled holes are not considered to be material form surfaces and the requirement for examination of holes (if any) resides in NX-4000 and NX-5000. The licensee concluded that the reexamination of machined surfaces as discussed in the ASME Code Section III, Article NB-4121.3 did not apply to the accessible interior surfaces of the flange stud holes because they were not material form surfaces.

On October 6, 2011, the NRC issued TIA No. 2011-15, which documented the Agency position on the application of the CC requirements. Specifically, the NRC determined that examination of the accessible surfaces of the RRVCH flange stud holes by MT or PT was required to meet the requirements of Articles NB-2541(a) and NB-4121.3 of Section III of the ASME Code. The licensee entered this issue into the corrective action system in multiple CRs (reference CR-2011-00344, CR-2011-01739 and CR 2011-04373) and subsequently completed MT examination of the accessible surfaces of the 60 RRVCH flange stud holes prior to placing the vessel head into service. At each stud hole, the accessible surface for MT examination included 3 inches in depth from the flange top and bottom surface and in total, amounted to an additional 7,917 square inches of surface area examined. No rejectable fabrication flaws were identified during this examination.

<u>Analysis</u>

The inspectors determined that failure to perform an adequate review of fabrication records to ensure material procured from a contractor (RRVCH) met the CC was contrary to 10 CFR 50, Appendix B, Criterion VII, and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of Equipment Performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Absent NRC identification, the licensee would not have completed surface examination of the 60 flange stud holes to ensure unacceptable material flaws (e.g., cracks) were not placed in service. Because material flaws such as cracks serve as stress risers that reduce the ability of the RRVCH to withstand failure by crack propagation during design basis events (e.g., pressurized thermal shock), they would place the reactor coolant system at an increased risk for through-wall leakage and/or failure. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Initiating Events Cornerstone. Because this finding was identified prior to placing the RRVCH in service and no fabrication flaws were identified, the inspectors answered "no" to the Significance Determination Process Phase 1 screening question "Assuming worst case degradation, would the finding result in exceeding the TS limit for any reactor coolant system leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?" Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Decision Making because the licensee staff failed to demonstrate that nuclear safety was an overriding priority in decisions affecting the RRVCH. Specifically, the failure to perform

an adequate review of the RRVCH fabrication records was caused by the licensee's decision to not review the manufacturer's interpretations and application of the CC rules (IMC 0310 – Item H.1.b). The inspectors reached this conclusion based on discussions with licensee staff and review of the licensee's apparent cause evaluation documented in CR-2011-04373.

Enforcement

Appendix B of 10 CFR 50, Criterion VII, "Control of Purchased Material, Equipment, and Services," requires in part that "Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents." And: "This documentary evidence shall be retained at the nuclear power plant, or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment."

Contrary to the above, as of July 22, 2011, the licensee had not established adequate measures (e.g. adequate review of vendor fabrication records) to ensure material procured from a contractor for the RRVCH conformed to the procurement documents. Specifically, licensee measures were not sufficient to ensure that surface examinations of 60 flange stud holes were completed in accordance with Section III of the ASME Code as required by Procurement Specification BUHSDB/NCC001 issued on January 22, 2002, and Purchase Order 7084643 issued on January 31, 2002. Because this violation was of very low safety significance and it was entered into the licensee's CAP (reference CR 2011-00344, CR 2011-01739, and CR 2011-04373), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000346/2011005-07)

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 10, 2012, the inspectors presented the inspection results to the Director of Site Operations, Mr. Brian Boles, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Radiological Hazard Assessment and Exposure Controls Program inspections under the Occupational Radiation Safety Cornerstone with the Site Vice President, Mr. Barry Allen, on October 21, 2011;
- Radiation Monitoring Instrumentation Program and Performance Indicator Verification under both the Public Radiation Safety Cornerstone and the Occupational Radiation Safety Cornerstone with the Site Vice President, Mr. Barry Allen, on September 16, 2011. Additionally, a telephone re-exit was conducted on October 21, 2011; and
- The Reactor Vessel Head Replacement Fabrication Review (IP 71007) with the Director of Special Projects, Mr. Clark Price, and other members of the licensee's staff on November 23, 2011.

- The Triennial Heat Sink Performance Review, the inspectors presented the inspection results to Mr. Barry Allen, and other members of the licensee staff, on January 31, 2012 via telephone conference. The licensee acknowledged the issues presented.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee</u>

B. Allen, Site Vice President
P. Boissoneault; Manager, Chemistry
B. Boles, Director, Site Operations
K. Byrd, Director, Site Engineering
T. Chowdhary, NRC Liaison
J. Dominy, Director, Site Maintenance
J. Hook, Manager, Design Engineering
R. Hovland, Manager, Training
G. Kendrick, Manager, Site Outage Management
P. McCloskey, Manager, Site Regulatory Compliance
D. Noble, Manager, Radiation Protection
W. O'Malley, Manager, Nuclear Oversight
R. Oesterle, Superintendent, Nuclear Operations
M. Parker, Manager, Site Protection
R. Patrick, Manager, Site Work Management
D. Petro, Manager, Steam Generator Replacement Project
S. Plymale, Manager, Site Operations
C. Price, Director, Special Projects
M. Roelant, Manager, Site Projects
C. Sacha, Radiation Protection Services Supervisor
D. Saltz, Manager, Site Maintenance
S. Steagall, Fleet Oversight Manager
C. Steenbergen, Superintendent, Operations Training
J. Stelmaszak, Supervisor of NSSS Plant Engineering
J. Sturdavant, Regulatory Compliance
T. Summers, Manager, Plant Engineering
L. Thomas, Manager, Nuclear Supply Chain
M. Travis, Superintendent, Radiation Protection
J. Vetter, Manager, Emergency Response
A. Wise, Manager, Technical Services
G. Wolf, Supervisor, Regulatory Compliance
K. Zellers, Supervisor, Reactor Engineering

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000346/2011005-01	NCV	Inadequate Control of Weld Filler Metal Electrodes (Section 1R08.1)
05000346/2011005-02	FIN	Decay Heat Pump 1-1 Damaged and Rendered Inoperable By Personnel Climbing on Equipment (Section 1R13.1)
05000346/2011005-03	NCV	Air Voids in Component Cooling Water System Caused By Inadequate Fill and Vent Procedure (Section 1R15.1)
05000346/2011005-04	NCV	Reactivity Manipulations Performed By Non-Licensed Individual (Section 1R19.1)
05000346/2011005-05	NCV	Inadequate Information on Valve Interlocks Resulted in Inadvertent Operation and Loss of Component Cooling Water Surge Tank Inventory (Section 1R20.1)
05000346/2011005-06	FIN	Inadequate Procedure Resulted in Water Intrusion Into Safety-Related Motor Control Center (Section 4OA3.2)
05000346/2011005-07	NCV	Incomplete Surface Examination of the RRVCH (Section 4OA5.3)

Closed

05000346/2011005-01	NCV	Inadequate Control of Weld Filler Metal Electrodes (Section 1R08.1)
05000346/2011005-02	FIN	Decay Heat Pump 1-1 Damaged and Rendered Inoperable By Personnel Climbing on Equipment (Section 1R13.1)
05000346/2011005-03	NCV	Air Voids in Component Cooling Water System Caused By Inadequate Fill and Vent Procedure (Section 1R15.1)
05000346/2011005-04	NCV	Reactivity Manipulations Performed By Non-Licensed Individual (Section 1R19.1)
05000346/2011005-05	NCV	Inadequate Information on Valve Interlocks Resulted in Inadvertent Operation and Loss of Component Cooling Water Surge Tank Inventory (Section 1R20.1)
05000346/2011004-01	URI	Plant Transient During HPI Flow Instrument String Checks (Section 4OA5.1)
05000346/2011005-06	FIN	Inadequate Procedure Resulted in Water Intrusion Into Safety-Related Motor Control Center (Section 4OA3.2)
05000346/2011005-07	NCV	Incomplete Surface Examination of the RRVCH (Section 4OA5.3)
05000346/2011004-05	URI	Code Surface Examination Requirements Not Applied to Closure Head Stud Holes (Section 4OA5.3)
05000346/2011-001-00	LER	Pressurizer Code Safety Valve Setpoint Test Failures (Section 4OA3.1)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

Condition Reports:
- 2011-06080; Technical Specification Freeze Point Reads All *****
- 2011-96566; Open Input Found on Freeze Circuit #90
- 2011-05352; Warehouse Repairs Required for ANSI N45.2.2 Storage Compliance
- 2011-04606; SH5968, Secondary Hot Water Heating Loop Recirculation Heat Exchanger Temperature Control Valve, Not Maintaining 185 Degrees Fahrenheit

Procedures:
- DB-OP-06913; Seasonal Plant Preparation Checklist; Revision 22
- DB-OP-06331; Freeze Protection & Electrical Heat Trace; Revision 20
- DB-ME-09521; Preventative Maintenance & Circuit Testing of Freeze Protection and Heat Tracing; Revision 4
- NOP-OP-1012; Material Readiness and Housekeeping Inspection Program; Revision 7
- NOP-WM-4006; Conduct of Maintenance; Revision 5

Work Orders:
- 200391823; PM 0912 Check Technical Specification Related Freeze Protection Heat Trace

1R04 Equipment Alignment

Condition Reports:
- 2011-01109; Decay Heat Pump 1 Outboard Axial Vibrations in Alert Range
- 2011-02969; Decay Heat 2733 Leakby Needs Quantified
- 2011-04403; DB-PF-03012 (DB-PF-03011) Emergency Core Cooling System Integrated Train 1 (2) Leakage Test Section 4.5 Decay Heat 2733 (Decay Heat 2734) not Performed on 24 Month Interval
- 2011-05687; Decay Heat Valve Pit Leak Test Does Not Meet Acceptance Criteria
- 2011-97140; Valves Added to Inservice Testing Program

Procedures:
- DB-SP-03019; Service Water Valve Verification Monthly Test Train 1; Revision 11
- DB-OP-06261; Service Water System Operating Procedure; Revision 47
- DB-OP-06012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 52
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 50

Drawings:
- OS-020, sheet 1; Service Water System; Revision 84
- OS-004, sheet 1; Decay Heat Removal/Low Pressure Injection System; Revision 50
- OS-041A, sheet 2; Emergency Diesel Generator Systems; Revision 29
- OS-041B, sheet 1; Emergency Diesel Generator Air Start/Engine Air System; Revision 40

1R05 Fire Protection

Conditions Report:
- 2011-04510, Compressed Gas Cylinders Secured with Scaffold Wire

Pre-Fire Plans:
- PFP-CB-410; East Elevation 603' and Valve Room Elevation 636', Rooms 410 and 580, Fire Area D; Revision 4
- PFP-CB-215; Let Down Coolers Area, Room 215, Fire Area D; Revision 5

Drawings:
- A-222F; Fire Protection, General Floor Plan El. 565'-0"; Revision 15
- A-223F; Fire Protection, General Floor Plan El. 585'-0"; Revision 21
- A-224F; Fire Protection, General Floor Plan El. 603'-0"; Revision 23
- A-225F; Fire Protection, General Floor Plan El. 623'-0"; Revision 18
- A-226F; Fire Protection, General Floor Plan El. 643'-0"; Revision 13

Other:
- Fire Hazard Analysis Report; Revision 24

1R07 Heat Sink Performance

Condition Reports:
- 2011-03664; Spent Fuel Pool Heat Exchanger 1 Does Not Meet Extrapolated Heat Transfer Rate
- 2011-03776; Procedure Improvements for Spent Fuel Pool Heat Exchanger Performance Test (DB-PF-04707)

Procedures:
- DB-PF-04707; Spent Fuel Pool Heat Exchangers; Revision 3

Drawings:
- OS-007; Spent Fuel Pool Cooling System; Revision 23

Work Orders:
- 200198502; Spent Fuel Pool Heat Exchanger Performance Test; 10/14/2011

1R07T Heat Sink Performance – Triennial

Condition Reports Generated as a Result of the Inspection:
- 11-00422; Intake Canal Dike Does Not Meet Design Configuration Requirements; August 10, 2011
- 11-97166; Degradation of the Intake Canal North Wall in the Q/NQ Portion of the Canal; June 30, 2011

Condition Reports:
- 09-62045; FME: Lost 1 Screw, 1 Nut and 2 Washers in CCW Train 1; July 28, 2009
- 10-79648; CCW hx 3 Repair per CR10-79648-CA6; July 14, 2010
- 10-80388; CCW hx 3 Returned to Service with 2 Areas Below Min Wall; July 28, 2010
- 10-81143; Three CCW hx 2 Tubes Remained Blocked After Cleaning Activities; August 12, 2010

- 10-83726; NRC Questions Process Control of CCW HX Straightening Process; October 5, 2010
- 11-87861; Flange Distortion From Weld Repair On No. 3 CCW HX; January 6, 2011
- 11-89424; CCW hx Unavailable > 30 Days Risk-Evaluation; February 10, 2011
- 11-89559; CCW hx 3 Straightening and Welding Issue @ 90 and 270; February 14, 2011
- 11-90322; Crack in Weld on No. 3 CCW HX During Repair; March 2, 2011
- 11-90364; Crack in Toe of Weld On No. 3 CCW HX; March 3, 2011
- 11-95719; Corrosion Identified in CCW HX 1; May 31, 2011
- 11-95924; Below Minimum Wall Thickness On CCW HX No.1 SW Side; June 3, 2011
- 11-96284; crack In Exterior Weld During Welding of CCW hx No. 1 SW Side; June 10, 2011
- 11-96398; CCW hx 1 Outer Tube Sheet to Channel Head Weld Crack; June 14, 2011
- 11-96432; Indications on Dye Penetrant Exam on No. 1 CCWHX; June 14, 2011
- 11-96441; Dye Penetrant Test Indications Found on No. 1 CCW hx; June 15, 2011
- 06-6749; CC1467 CCW from Decay Heat Cooler 1 Solenoid Outlet Valve Would not Close; September 24, 2006
- 08-50956; Packing Loads on AF3869 Exceed Current Design Values; December 16, 2008
- 09-64856; Containment Spray Pump No. 1 "as found" Information; September 23, 2009
- 10-69758; AF 68 Check Valve Failure; January 9, 2010
- 11-88345; SFAS Ch. 3 CTMT Pressure Test Switch Required Agitation When Released From Test; January 17, 2011
- 11-90425; EDG Exhaust Missile Barrier Grating Attachment Discrepancies; March 4, 2011
- 11-00422; Intake Canal Dike Does Not Meet Design Configuration Requirements; August 10, 2011
- 11-97166; Degradation of the Intake Canal North Wall in the Q/NQ portion of the Canal; June 30, 2011
- 11-95451; S33-2DB-SS-03711; CTRM Emerg Vent Sys Train 2 Performance Test; May 27, 2011
- 07-26185; Degradation on Rip-Rap Sides of the FOREBAY and Intake Canal; August 9, 2007
- 09-54330; Slope Stability Study for the FOREBAY North Wall Found Low Strength Clay Till; February 27, 2009

Procedures:
- DB-OP-06261; Service Water System Operating Procedure; Revision 45
- DB-OP-02011; Heat Sink Alarm Panel 11 Anunciators ; Revision 09
- NORM-OP-1009; SRO Review of Condition Reports; Revision 00
- NOP-LP-2001; Corrective Action Program; Revision 27
- DB-OP-03007; Miscellaneous Instrument Daily Checks; Revision 19
- RA-EP-02820; Emergency Plan Off Normal Occurrence Procedure (Earthquake); Revision 07
- PM 8369; Inspection – Embankment Intake Canal; June 21, 2011
- PM 2694; Inspection – Intake Crib; June 21, 2011

Calculations:
- C-ICE-009.01-002; Ultimate Heat Sink Level; July 3, 2003
- 12501-703; Thermal Performance Analysis for UHS Pond; Revision 01
- 12501-M-004; Thermal Performance Analysis For UHS Station 0+00 – 10+00; Revision 01
- 12501-M-001; Thermal Performance Analysis For UHS Station 10+00; Revision 01

Other:
- 7749-M-23; CCW HX specification sheets; August 1, 1977
- Serial 1-823; Licensee Response to Bulletin 88-04 “Potential Safety-Related Pump Loss”; September 8, 1988

- 06-003; Standing Order for Limit on UHS; CREVS Train 1, Design/Licensing Basis; August 24, 2006
- 144188-0209-1575; Slope Stability Study, (Bowser Morner); February 11, 2009
- Serial 2654; Response to RAI LA 96-0008 (UHS Temp Increase); June 6, 2000
- RAS-00-00250; TeleCon NRC-FENOC LA 96-0008 (UHS Temp Increase);
- March 28-30, 2000
- Serial 2397; LAR 96-0008 (UHS Temp Increase); July 18, 2000
- RAS-98-00063; FENOC Presentation to NRR (License Basis and Design Basis of the UHS); February 17, 1998
- Serial 2347; Ultimate Heat Sink/Service Water Temperature; January 31, 1996
- 2011-01; UHS Limitation Due to CREATCS Calc Issue (CR 11-95467); Revision 00
- 154381-0811-2995; Intake Canal Study, (Bowser Morner); August 05, 2011

Work Orders:
- WO 200426314; Completed Repairs on CCW HX 3 per CR 10-80388; June 13, 2011
- WO 200423055 Addendum 1; Completed Repairs on ccw hx 3 per CR10-79648-CA6; June 13, 2011
- WO 200117165; Tubes Plugged in ccw hx 2; September 20, 2010
- WO 200284546; Work Order for Intake Canal Fix; October 5, 2007
- WO 200220528; Intake Canal Inspection; May 29, 2007
- WO 200325296; CCW HX 3 Performance Test Completed July 9, 2010; Revision 08

Audits, Assessments and Self-Assessments:
- SN-SA-11-191; Snapshot Assessment of Reliability of Heat Exchangers Cooled by Service Water and GL 89-13 Implementation; June 16, 2011

Tests:
- DB-PF-04706 Order 200325296; CCW HX 3 Performance Test Completed July 9, 2010; Revision 8

Modifications:
- C-ICE-009.01-002 A01; Ultimate Heat Sink Level; Multiplexor Replacement; February 23, 2010
- 07-00464; CAC Heat Duty at Elevated SW Inlet Temperature; January 31, 2007

Drawings:
- 7749-M-23-8-5; Struther Wells Corporation CCW HX Details Drawing; December 6, 1971
- C-1; Site Plan; Revision 21
- OS-032B; Control Room Emergency Ventilation System; Revision 18
- OS-020; SH 2 Operational Schematic Service Water System; Revision 42
- C-49; Discharge System; Revision 21
- C-46; Discharge System; Revision 28
- C-45; Flood Control Dike-Sections; Revision 11
- C-45A; Flood Control Dike-Section and Intake Structure Miscellaneous Steel; Revision 01
- C-45B; Intake Structure Miscellaneous Steel Sections and Details; Revision 00
- C-45C; Intake Structure Miscellaneous Steel Sections and Details; Revision 00
- TEC 201 B1; Water Intake and Discharge; Sh. 3 of 17 (Finkbeiner, Drawing); December 20, 1972

<u>1R08 Inservice Inspection</u>

Condition Reports:
- 2011-94103; SFP Pump 1-2 Seal Leak; dated May 4, 2011
- 2011-05605; NRC Question of 17M RPV UT Data (S-dim); dated November 17, 2011
- 2011-05118; Inadequate Reconciliation of E22-3 HX Repair; dated November 4, 2011
- 2011-04942; Results of BMV of the RVCH; dated November 4, 2011
- 2011-04891; NRC UT Procedure Issue; dated November 3, 2011
- 2011-04847; Reactor Head BMV Procedure Briefing; dated November 3, 2011
- 2011-04810; Incomplete Quality Record; Dated November 2, 2011
- 2011-03984; Failure to Declare Component Inoperable Due to ISI Indication; dated October 19, 2011
- 2011-03875; Results of ISI Exam of the Exterior Containment Vessel Moisture Barrier; dated October 17, 2011
- 2011-02113; Linear Indication HP92 Base Metal; dated September 19, 2011
- 2010-79648; Corrosion at T-Weld for CCW HX 1-3; dated June 26, 2010
- 2010-79012; HPI Pump 2P58-2 Boric Acid at Mechanical Seal; dated June 29, 2010
- 2010-78548; RCS to DH System Leak at DH22A Packing; dated June 20, 2010
- 2010-78373; OA for EOC 17; dated June 16, 2010
- 2010-77851; Debris on Lower Core Grid; dated June 6, 2010
- 2010-76667; Packing Leak SF 35; dated May 10, 2010
- 2010-75053; Debris at FW51; dated April 8, 2010
- 2010-74892; RCP 1-2 Boric Acid Leak at Bolted Connection; dated April 5, 2010
- 2010-73653; DH11 Packing Leak; dated March 16, 2010
- 2010-73412; 16 RFO Debris on Fuel Assembly; dated March 14, 2010

Miscellaneous Documents:
- NIS-2 Form; CCW HX 1-3; dated July 23, 2010
- Welding and NDE Services Lab Work Request; dated July 26, 2011
- Davis-Bessel Project Welder Qualification List; dated October 17, 2011
- Non Destructive Examination Records
- Liquid Penetrant Examination Report 17-PT-011; Valve HP92 to Pipe Weld; dated September 19, 2011
- Liquid Penetrant Examination Report 17-PT-012; Valve HP92 to Pipe Weld; dated September 19, 2011
- Liquid Penetrant Examination Report 17-PT-036; MU-31-CCA-18-1-FW23; dated October 13, 2011
- UT Calibration/Examination Report 17-UT-053; RC-PZR-WP-15; dated October 12, 2011
- Reactor Vessel 10 Year Ultrasonic Examination Summary Report; dated October 20, 2011

Procedures:
- CR-ASME III CL B; Nondestructive Examination Standard Computed Radiographic Examination; Revision 0
- 54-ISI-367-11; Visual Examination for Leakage of Reactor Head Penetrations; dated January 26, 2010
- 54-ISI-69-31; Administrative Procedure for Processing Nondestructive Examination Data; dated February 26, 2010
- 54-PT-200-04; Color Contrast Solvent Removable Liquid Penetrant Examination of Components; dated June 24, 2011
- 54-MT-02-08; Wet or Dry Magnetic Particle Examination Procedure; dated February 24, 2009

- 54-ISI-801-02; Automated Ultrasonic Examination of PWR Vessel Shell Welds; dated February 14, 2011
- 54-ISI-805-07; Ultrasonic Examination of Reactor Pressure Vessel Welds; Revision 7
- NOP ER-2001; Boric Acid Corrosion Control Program; Revision 9
- NOP DP-01501; Boric Acid Corrosion Control Inspection; Revision 13
- Weld Procedures and Qualification Records:
- GWS Arc Welding Standard Arc Welding of Reinforced Steel (GWS-Rebar), Revision 0
- WPS P1-Rebar (Indiret-0.57 CE); Revision 0
- WPS P1-Rebar (0.87 CE); Revision 0
- WPS P1-A-Lh(CVN 0F); Revision 1
- PQR 1310; dated June 20, 2001
- PQR 1675; dated August 12, 2011
- PQR 1359 P1-Rebar (Indiret-0.57 CE); dated June 18, 2003
- PQR 1392 P1-Rebar (0.87 CE); dated September 30, 2003

Welder Qualification Records:
- WR-1 Welder Performance Qualification Test Record- CBI-3; dated September 13, 2011
- WR-1 Welder Performance Qualification Test Record- CBI-7; dated September 13, 2011
- WR-1 Welder Performance Qualification Test Record- CBI-13; dated October 2, 2011
- WR-1 Welder Performance Qualification Test Record- CBI-14; dated October 2, 2011
- WR-1 Welder Performance Qualification Test Record- I-1; dated October 7, 2011
- WR-1 Welder Performance Qualification Test Record- I-4; dated October 10, 2011

Weld Data Records:
- Madison Inc Certified Material Test Report-A516 Grade 70 Plate - Heat 1505681; dated September 19, 2010
- WR-5C Structural Field Welding Checklist- FW13 and 14; dated October 24, 2011
- WR-5 Field Welding Checklist- FW-1; dated October 22, 2011
- WR-6 Filler Metal Withdrawal Form E 7018; dated October 23, 2011
- WR-6 Filler Metal Withdrawal Form E 7018; dated October 22, 2011
- WR-6 Filler Metal Withdrawal Form E 7018; dated October 21, 2011
- ELAB Group Inc CMTR No. 743965-E9018-B3H4R- Heat M100021; dated January 21, 2011
- GERDAU CMTR Heat No. K115877- No. 10 Rebar; dated September 1, 2011
- GERDAU CMTR Heat No. K111485- No. 11 Rebar; dated September 1, 2011
- GERDAU CMTR Heat No. C014692- No. 8 Rebar; dated September 1, 2011
- GERDAU CMTR Heat No. J112953- No. 10 Rebar; dated September 1, 2011
- Mistras CE Rebar Sample Chemical Analysis; dated November 3, 2011

1R12 Maintenance Effectiveness

Condition Reports:
- 2008-33710; Groundwater In-seepage Identified In The Annulus Sandpocket
- 2011-01540; Exterior Shield Building Inspection Findings
- 2011-03346: Fractured Concrete Found at 17M Shield Building Construction Opening
- 2011-03875; Results of ISI Examination of the Exterior of the Containment Vessel Moisture Barrier
- 2011-03996; Extent of Condition for Shield Building Fracture Indications
- 2011-04190; Surface Cracks Identified on Fluted Areas of the Shield Building
- 2011-04214; Core Bore Found Additional Crack in Architectural Flute Area
- 2011-04402; Fractured Concrete Found at 17M Shield Building at Main Steam Line Penetrations

- 2011-04507; Isolated Crack Indication Identified by Impulse Response Testing
- 2011-04648; Shield Building IR Indications above Elevation 780
- 2011-05475; Concrete Cracking at the Top of the Shield Building Wall
- 2011-05648; Concrete Cracking in Shoulder 4 / Flute 2 Region of the Shield Building (Azimuth 67.5)
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06185; Error in Calculation C-CSS-099.20-056, Revision 01

Procedures:
- NOP-ER-3004; FENOC Maintenance Rule Program; Revision 01
- EN-DP-01511; Design Guidelines For Maintenance Rule Evaluation of Structures; Revision 0
- DB-PF-03009; Containment Vessel and Shield Building Visual Inspection; Revision 7

Other:
- MRPM; Maintenance Rule Program Manual; Revision 29
- Maintenance Rule Unavailability Hours Database
- Maintenance Rule Evaluation Worksheets; Containment Shield Building Dome; dated September 2, 2005 and June 14, 1999
- Maintenance Rule Evaluation Worksheets; Containment Shield Building Exterior; dated October 17, 2005 and June 14, 1999
- Maintenance Rule Evaluation Worksheets; Containment Shield Building Interior; dated May 3, 2010 and January 2, 2008
- Maintenance Rule Evaluation Worksheets; Containment Vessel Exterior; dated May 3, 2006 and May 4, 1998
- Maintenance Rule Evaluation Worksheets; Containment Vessel Interior; dated April 1, 2010, May 1, 2010, January 23, 2008, March 17, 2007 and May 12, 1998

1R13 Maintenance Risk Assessments and Emergent Work Control

Condition Reports:
- 2011-03022; DB-PA-11-03: Configuration Control Issue Noted During RCS Drain
- 2011-03346: Fractured Concrete Found at 17M Shield Building Construction Opening
- 2011-03465; DB-PA-11-03: Issues Identified With Containment Closure Documentation and Implementation
- 2011-03996; Extent of Condition for Shield Building Fracture Indications
- 2011-04190; Surface Cracks Identified on Fluted Areas of the Shield Building
- 2011-04214; Core Bore Found Additional Crack in Architectural Flute Area
- 2011-04402; Fractured Concrete Found at 17M Shield Building at Main Steam Line Penetrations
- 2011-04507; Isolated Crack Indication Identified by Impulse Response Testing
- 2011-04648; Shield Building IR Indications above Elevation 780
- 2011-05475; Concrete Cracking at the Top of the Shield Building Wall
- 2011-05648; Concrete Cracking in Shoulder 4 / Flute 2 Region of the Shield Building (Azimuth 67.5)
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06185; Error in Calculation C-CSS-099.20-056, Revision 01
- 2011-07195; DH Pump 1-1 O/B Bearing Oil Temp Element Damaged

Procedures:
- NOP-OP-1007; Risk Management; Revision 9
- DB-MM-09234; Equipment Hatch Removal and Reinstallation; Revision 8

- NG-DB-00117; Shutdown Defense In Depth Assessment; Revision 11
- DB-OP-06904; Shutdown Operations; Revision 37
- DB-OP-1005; Shutdown Defense In Depth; Revision 13
- NOP-OP-1002; Conduct of Operations; Revision 5

Business Practices:
- DBBP-OPS-0003; On-Line Risk Management Process; Revision 10
- DBBP-OPS-0011; Protected Equipment Posting; Revision 3

Calculations:
- C-NSA-099.16-023; Risk Significant Component Matrix – Attachment 7; Revision 7
- C-CSS-099.20-046; Evaluation of Shield Building for the Permanent Condition; Revision 0
- C-CSS-099.20-047; Restoration of Shield Building Construction Opening; Revision 0
- C-CSS-099.20-053; Evaluation of Shield Building for the Interim Condition with Outside Vertical Reinforcement Removed at Each Flute Shoulder; Revision 0
- C-CSS-099.20-054; Evaluation of Shield Building for the Permanent condition with Outside Vertical Reinforcement Removed at Cracking Areas; Revisions 0, 1, 2, and 3
- C-CSS-099.20-055; II/I Evaluation for Architectural Flute Shoulder; Revision 0
- C-CSS-099.20-056; Evaluation of Shield Building Hoop Reinforcement with Observed Cracking; Revisions 0 and 1

Drawings:
- C-111A; Shield Building Exterior Developed Elevation; Revision 0 and 1

Other:
- MRPM; Maintenance Rule Program Manual; Revision 29
- 17M Shutdown Defense In Depth Report; Revision 1
- Davis-Besse Shield Building Investigation and Technical Summary; Revisions 0 and 1
- Davis-Besse Shield Building Cracking Investigation and Assessment Report; Revisions 0 and 1

1R15 Operability Evaluations

Condition Reports:
- 2011-01902; Extent of Condition Concerns from CR 2011-98223
- 2011-03346: Fractured Concrete Found at 17M Shield Building Construction Opening
- 2011-03996; Extent of Condition for Shield Building Fracture Indications
- 2011-04190; Surface Cracks Identified on Fluted Areas of the Shield Building
- 2011-04214; Core Bore Found Additional Crack in Architectural Flute Area
- 2011-04402; Fractured Concrete Found at 17M Shield Building at Main Steam Line Penetrations
- 2011-04507; Isolated Crack Indication Identified by Impulse Response Testing
- 2011-04648; Shield Building IR Indications above Elevation 780
- 2011-05475; Concrete Cracking at the Top of the Shield Building Wall
- 2011-05648; Concrete Cracking in Shoulder 4 / Flute 2 Region of the Shield Building (Azimuth 67.5)
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06185; Error in Calculation C-CSS-099.20-056, Revision 01
- 2011-98223; DC System Issues from NRC CDBI
- 2011-05526; Acceptance Criteria For Train 2 Service Water Flow Balance, DB-SP-03001, Not Met

- 2011-05283; DB-SP-03001 Service Water Loop 2 Integrated Flow Balance Procedure Flow Does Not Meet Minimum Acceptance Criteria
- 2011-05163; Questionable Margin of CCW SW Flow Balance Acceptance Criteria
- 2011-05183; Cycling of #2 SW Pump Strainer During Performance of DB-SP-03001

Procedures:
- NOP-LP-2001; Corrective Action Program; Revision 27
- NOP-OP-1009; Operability Determinations and Functionality Assessments; Revision 3
- NOP-OP-1014; Plant Status Control; Revision 1
- NOBP-OP-0004; Plant Status Control and Clearance Events; Revision 4
- DB-OP-02011; Heat Sink Alarm Panel 11 Annunciators; Revision 9
- DB-OP-02521; Loss of AC Bus Power Sources; Revision 17
- DB-SP-03001; Service Water Loop 2 Integrated Flow Balance Procedure; Revision 15
- DB-CH-06033; Auxiliary Systems Chemical Addition; Revision 19

Calculations:
- C-EE-001.01-010; DC Calc-Battery/Charger Size, Short Circuit, Voltage Drop; Revision 31
- C-CSS-099.20-046; Evaluation of Shield Building for the Permanent Condition; Revision 0
- C-CSS-099.20-047; Restoration of Shield Building Construction Opening; Revision 0
- C-CSS-099.20-053; Evaluation of Shield Building for the Interim Condition with Outside Vertical Reinforcement Removed at Each Flute Shoulder; Revision 0
- C-CSS-099.20-054; Evaluation of Shield Building for the Permanent condition with Outside Vertical Reinforcement Removed at Cracking Areas; Revisions 0, 1, 2, and 3
- C-CSS-099.20-055; II/I Evaluation for Architectural Flute Shoulder; Revision 0
- C-CSS-099.20-056; Evaluation of Shield Building Hoop Reinforcement with Observed Cracking; Revisions 0 and 1
- C-NSA-011.01-016; Service Water System Design Basis Flowrate Analysis and Testing Requirements

Work Orders:
- 200379719; Service Water Train 2 Flow Balance Simple Troubleshooting

Drawings:
- E-0007; 250/125 Vdc and Instrumentation AC One Line Diagram; Revision 40
- E-642A, Sheet 1A; Nonessential 125 Vdc Distribution Panel “DAP” Channel – A; Revision 13
- E-642A, Sheet 1B; Nonessential 125 Vdc Distribution Panel “DAP” Channel – A; Revision 15
- E-642A, Sheet 2A; Nonessential 125 Vdc Distribution Panel “DBP” Channel – B; Revision 15
- E-642A, Sheet 2B; Nonessential 125 Vdc Distribution Panel “DBP” Channel – B; Revision 15
- E-642A, Sheet 3A; Nonessential 125 Vdc Distribution Panel “DAN” Channel – A; Revision 10
- E-642A, Sheet 3B; Nonessential 125 Vdc Distribution Panel “DAN” Channel – A; Revision 5
- E-642A, Sheet 4A; Nonessential 125 Vdc Distribution Panel “DBN” Channel – B; Revision 11
- E-642A, Sheet 4B; Nonessential 125 Vdc Distribution Panel “DBN” Channel – B; Revision 8
- E-640A, Sheet 1A; Essential 125 Vdc Distribution Panel “D1P” Channel – 1; Revision 22
- E-640A, Sheet 1B; Essential 125 Vdc Distribution Panel “D1P” Channel – 1; Revision 14
- E-640A, Sheet 2A; Essential 125 Vdc Distribution Panel “D2P” Channel – 2; Revision 21
- E-640A, Sheet 2B; Essential 125 Vdc Distribution Panel “D2P” Channel – 2; Revision 14
- E-640A, Sheet 3A; Essential 125 Vdc Distribution Panel “D1N” Channel – 3; Revision 11
- E-640A, Sheet 3B; Essential 125 Vdc Distribution Panel “D1N” Channel – 3; Revision 3
- E-640A, Sheet 4A; Essential 125 Vdc Distribution Panel “D2N” Channel – 4; Revision 12
- E-2013H, Sheet No. 1 of 12; Station 125 Vdc Distribution System Failure Analysis Manual Cover Sheet Introduction; Revision 2

- E-2013J, Sheet No. 1 of 71; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel D1P; Revision 3
- E-2013K, Sheet No. 1 of 118; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel DAP; Revision 3
- E-2013L, Sheet No. 1 of 8; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel D1N; Revision 2
- E-2013M, Sheet No. 1 of 38; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel DAN; Revision 2
- E-2013N, Sheet No. 1 of 68; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel D2P; Revision 3
- E-2013P, Sheet No. 1 of 103; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel DBP; Revision 3
- E-2013Q, Sheet No. 1 of 7; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel D2N; Revision 3
- E-2013R, Sheet No. 1 of 43; Station 125 Vdc Distribution System Failure Analysis Manual – Cover Sheet Panel DBN; Revision 3
- OS-060, Sheet 1; Operational Schematic 250/125 Vdc and 120 V Instrument AC System; Revision 17
- OS-060, Sheet 2; Operational Schematic 250/125 Vdc and 120 V Instrument AC System; Revision 17
- C-111A; Shield Building Exterior Developed Elevation; Revision 0 and 1

Other:
- Davis-Besse Shield Building Investigation and Technical Summary; Revisions 0 and 1
- Davis-Besse Shield Building Cracking Investigation and Assessment Report; Revisions 0 and 1

1R18 Plant Modifications

Condition Reports:
- 2011-02133; ICS/ULD Upgrade HLG NR TEMP Divergence Alarms
- 2011-02136; ICS/ULD Upgrade Communication Alarm to PPC
- 2011-02253: Penetration Sleeve Size Error Resulted in Nonconservative Analysis Error
- 2011-02261; Unanalyzed Failure Mode of ICS/ULD Modification
- 2011-03125; Inadvertent Short in NNI X
- 2011-04311; Service Water Pipe Project – Jacking Bolt Cannot Be Installed In Orifice Flange
- 2011-04267; Service Water Supports A-328 and A-341 Do Not Match Drawing Offsets
- 2011-04491; SW Pipe 3" –HABC-45 Not Installed Per Drawings by WSI Under Order 200432711
- 2011-03418; Service Water Replacement Piping Tie-In As Found Conditions
- 2011-05689; As Found Data Out of Tolerance – Order 204428777 – ICS ULD & V Buffer FYSP02A1-D Found Failed
- 2011-95917; ICS ECP 02-0540-00 Makeup Flow Out of Range High in New ICS Unit Load Demand

Work Orders:
- 200432023; Replace SW Piping to ECCS Room Coolers
- 200316618; Replace SW ECCS Supply/Return Piping
- 200432711; Replace SW Piping to ECCS Room Cooler #1
- 200432714; Replace SW Piping to ECCS Room Cooler #2
- 200432715; Replace SW Piping to ECCS Room 1 Coolers

- 200432716; Replace SW Piping to ECCS Room Cooler #4
- 200432717; Replace SW Piping to ECCS Room Cooler #5
- 200432718; Replace SW Piping From ECCS Room 2 Coolers

Procedures:
- NOP-OP-4106; Control of Radiography Operations; Revision 2

Engineering Change Packages:
- 02-0540-000; Integrated Control System (ICS) Unit Load Demand Replacement
- 10-0458-000; SGR – 17M – Install Shield Building Construction Opening
- 10-0459-003; SGR – 17M – Containment Vessel Opening, Cut Wall Opening
- 10-0459-004; SGR – 17M – Containment Vessel Opening, Restore Wall
- 11-0412-000; Removal of ECCS Room Cooler Check Valves
- 11-0412-003; 17M Service Water Replacement

Other:
- USAR Section 9.2.1; Service Water System

<u>1R19 Post Maintenance Testing</u>

Condition Reports:
- 2008-33814; Diagnostic Testing on Valve SP6B
- 2011-03878; #1 Decay Heat Pump (P42-1) Bearing Housing Has Loose/Chipping Internal Coating
- 2011-04176; Incorrect Sealant Used on the Containment Air Cooler (CAC) #3 Endbell
- 2011-04244; CTMT Spray Baseline Test, Motor Data Greater Than 100 Percent FLA and Line Voltage Greater Than 110 Percent
- 2011-04251; Containment Spray (CS) Pump 1 Baseline Test Results
- 2011-04252; Inadequate Acceptance Criteria in DB-PF-03472, Makeup Pump 1 Baseline Test
- 2011-04338; #3 CTMT Air Cooler Motor Found Damaged
- 2011-04344; MP37-1A Makeup Baseline Test, Motor Data Greater Than 100 Percent FLA and Line Voltage Greater than 110 Percent
- 2011-04400; Battery Charger DBC1N – Termination of 500 MCM Cable at D134
- 2011-04437; Test Deficiency: Battery Charger DBC1N Low AC Voltage Disconnect Did Not Function
- 2011-04501; Question on the Configuration of the Condensate Drain Plug for CTMT Air Cooler Motor #3
- 2011-04574; MP42-1 Decay Heat Pump 1-1 Baseline Test (DB-PF-03236) Line Voltage Greater than 110 Percent of the Motor Nameplate Voltage and to Evaluate the Motor and Pump Hydraulic Data from DB-PF-03236
- 2011-04591; SP6B Has In-Body Thread Damage
- 2011-04602; At Step 6.5.34 of DB-SC-10023, DBC1P AC Input Breaker Did Not Trip When AC Power Was Removed (BE 1233 Opened)
- 2011-04620; Test Deficiency: Battery Charger DBC1P
- 2011-04830; Abnormal Thermal Indications Noted Via Infrared Thermography within BE1501
- 2011-04901; Solenoid Valves Required Rebuild After Maintenance
- 2011-04993; Data Recorded For DB-PF-03236 DH Pump 1 Baseline Testing
- 2011-05228; ILRT Procedure (DB-PF-10310) Discrepancy in Attachment 3E
- 2011-05630; Decay Heat Pit Has a Crack in the caulking That May Have Been Caused By/During the ILRT
- 2011-05687; Decay Heat Valve Pit Leak Test Does Not Meet acceptance Criteria

- 2011-05693; Relative Position Indication for Control Rod 2-2 Not Responding
- 2011-05716; Penetration Boot Seal Could Not Be Performed By Procedure
- 2011-05847; EVS Train 1 Refueling Interval SFAS Drawdown Test, DB-SS-03254, Failure
- 2011-06302; Auxiliary Feedwater Pump Turbine 1 Steam/Water Casing Leakage
- 2011-06318; Trip Keylock Switch to Insert Control Rod Groups During Testing Was Not Operated by a Licensed Individual

Procedures:
- DB-PF-03437; Containment Spray Pump 1 Baseline Test; Revision 3
- DB-PF-05000; Motor Testing; Revision 3
- DB-PF-05064; Electrical Machine Testing Using PdMA Motor Tester; Revision 9
- DB-SP-03157; AFP 1 Response Time Test; Revision 18
- DB-SP-03299; Containment Air Cooling Unit 3 18 Month Test; Revision 8
- DB-SP-03296; Containment Air Cooling Unit 3 Monthly Test; Revision 11
- DB-PF-03472; Makeup Pump 1 Baseline Test; Revision 4
- DB-PF-03236; Decay Heat Pump 1 Baseline Test; Revision 7
- DB-PF-10310; Containment Integrated Leakage Rate Test; Revision 7
- DB-ME-03003; Station Battery Charger Test; Revision 12
- DB-SC-03270; Control Rod Assembly Insertion Time Test; Revision 11
- DB-SC-03272; Control Rod Exercising Test; Revision 4
- DB-SC-10023; Post-Modification Test For Battery Charger DBC1P; Revision 0
- DB-SC-10024; Post-Modification Test For Battery Charger DBC1N; Revision 0
- DB-SS-03254; Emergency Ventilation System Train 1 Refueling Interval SFAS Drawdown Test; Revision 12
- DB-SC-03270; Control Rod Assembly Insertion Time Test; Revision 11
- NOP-WM-3620; Air Operated Valve Diagnostic Testing; Revision 0
- DB-OP-02011; Heat Sink Alarm Panel 11 Annunciators; Revision 9

Work Orders:
- 200005040; Decay Heat 1: Inspect/Recoat Bearing Housings. Replace Mechanical Seals
- 200376148; Makeup Pump 1 I/B Cover to Casing Gasket Replace
- 200389986; DBC1N Replace Charger ECP 02-0707-003
- 200389987; DBC1P Replace Charger ECP 02-0707-002
- 200404473; Makeup Pump 1 Replace O/B Seal
- 200423169; PM 6095 MC1-3 Motor Testing CAC Fan 3 Motor
- 200423797; ECP 10-0578 PM 7739 Makeup Pump 1 Refurb/Replace/Rewind Motor
- 200423980; ECP 11-0467 PM 9896 Decay Heat 1 Refurb/Rewind/Replace Motor
- 200423981; Containment Spray 1 PM 9897, Refurb/Rewind/Replace Motor
- 200427977; Rod Drop CRA Insertion Time Test
- 200428009; AFP 1 Response Time Test
- 200428641; PM 4301 FVSP6B Rebuild #1 MN FW
- 200429509; Containment ILRT
- 200437806; ECP 11-0143-001 CAC Fan #3 Motor Needs Replaced

Other:
- Containment Spray Pump 1 Motor Data, dated October 24, 2011
- Makeup Pump 1 Motor Data, dated October 25, 2011
- ECP 02-0707-002; Replace Battery Charger DB-DBC1P
- ECP 02-0707-003; Replace Battery Charger DB-DBC1N
- ECP 11-0143-001; Replacement Motor For MC1-3; Revision 0

1R20 Refueling and Other Outage Activities

Condition Reports:
- 2011-04881; Violation of NOP-LP-1202, Vehicle Found With Keys in Ignition
- 2011-04886; Requirements to Enter Mode 6 for Containment Closure After Vessel Opening
- 2011-05036; Paint Thickness Requirements Outside Specification
- 2011-05173; P78A Containment Normal Sump Pump A Did Not Meet Acceptance Criteria – Failed Test
- 2011-05419; Evaluate Areas in Containment Not Coated Per Specification A-024Q
- 2011-05517; Protective Coating Applied Inside Containment Not in Compliance With Order or Requirements of Engineering Change Package
- 2011-05558; POD 10-001 Review of Mode Change or Plant Operating Restrictions
- 2011-05588; Entry Into Plant Condition Prior to Resolving Listed Restraints
- 2011-05630; Decay Heat Pit Has a Crack in the caulking That May Have Been Caused By/During the ILRT
- 2011-05652; Pressurizer Insulation Not Properly Secured
- 2011-05672; Containment Walkdown
- 2011-05675; Peeling Paint and Tape in Containment
- 2011-05687; Decay Heat Valve Pit Leak Test Does Not Meet acceptance Criteria
- 2011-05690; Containment Unacceptable Items Found During Containment Closure Walkdown
- 2011-05692; Potential Rework for CRD Rod 2-2
- 2011-05709; Containment Closeout Inspection Tour With NRC Inspectors
- 2011-05711; Removed Outer Jacket From Sealtite in Containment
- 2011-05740; Mirror Insulation Clips Damaged, Missing, and Not Clipped
- 2011-05777; Void in Concrete at the top of the Shield Building pour back (Bechtel NCR 20)
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06196; Rising Reactor Coolant Drain Tank Level
- 2011-06291; Pressurizer Safety Valve Performance During Plant Heatup

Procedures:
- NG-DB-00117; Shutdown Defense in Depth Assessment; Revision 11
- NOP-OP-01005; Shutdown Defense in Depth; Revision 13
- DB-OP-06002; RCS Draining and Nitrogen Blanketing; Revision 19
- DB-OP-06003; Pressurizer Operating Procedure; Revision 28
- DB-OP-06230; Steam Generator Secondary Side Fill, Drain, and Layup; Revision 14
- DB-OP-06301; Generator and Exciter Operating Procedure; Revision 23
- DB-OP-06903; Plant Cooldown; Revision 40
- DB-OP-06904: Shutdown Operations; Revision 38
- DB-OP-06912; Approach to Criticality; Revision 16
- DB-MM-04004; Station Cranes Periodic Test; Revision 12
- DB-MM-06002; Polar Crane Operation; Revision 16
- DB-MM-09242; P&H station Crane Maintenance; Revision 2
- DB-MS-04009; Hand and Electric Operated Hoists Inspection; Revision 6
- DB-ME-09500; Installation and Termination of Electrical Cables; Revision 23

Business Practices:
- DBBP-ESAF-1010; Containment Cranes; Revision 0

Work Orders:
- 200429982; Polar Crane Inspections

Other:
- Bechtel Radiography Management Plan, Davis-Besse Containment Vessel, Fall 2011; Revision 0

1R22 Surveillance Testing

Condition Reports:
- 2011-03416; DB-SC-03121: Non-Conservative Computer Pt Used as Time Reference for Response Time Calcs
- 2011-04315; Required CR for IST Valve Times From 2011 SFAS Integrated Train 2 DB-SC-03121
- 2011-04935; EDG 2 Frequency High Test Deficiency in DB-SC-03121, Integrated SFAS Train 2
- 2011-05228; ILRT Procedure (DB-PF-10310) Discrepancy in Attachment 3E
- 2011-05630; Decay Heat Pit Has a Crack in the caulking That May Have Been Caused By/During the ILRT
- 2011-05687; Decay Heat Valve Pit Leak Test Does Not Meet Acceptance Criteria
- 2011-05716; Penetration Boot Seal Could Not Be Performed By Procedure
- 2011-06382; During Normal Operating Pressure, Normal Operating Temperature Walkdown Inspection the L7 Removable Insulation Panel on the Integrated Head Assembly could not be Extracted from the Installed Location
- 2011-88100; Pressurizer Code Safety Valves Setpoint Test Failure Reporting

Procedures:
- DB-PF-03008; Containment Local Leakage Rate Tests; Revision 15
- DB-SC-03121; SFAS Train 2 Integrated Response Time Test; Revision 2
- DB-SC-03074; Emergency Diesel Generator 1, ABDC1, and AC103 Appendix R Test; Revision 6
- DB-SP-03157; AFP 1 Response Time Test; Revision 18
- DB-PF-03010; RCS Leakage Test; Revision 11
- DB-PF-10310; Containment Integrated Leakage Rate Test; Revision 7

VT-2 Summary Numbers:
- B15.000.RC01; Primary Reactor Coolant System Class 1 Pipe for Leakage Test; 17-VT-286, 17-VT-287, 17-VT-288, 17-VT-289, 17-VT-290, 17-VT-291, 17-VT-292, 17-VT-293, 17-VT-294, 17-VT-295, 17-VT-297
- B15.000.DH15; Reactor Coolant System to Decay Heat Pump Suction (Bypass); 17-VT-298
- B15.000.DH14; Reactor Coolant System to Decay Heat Pump Suction (Normal); 17-VT-299
- B15.000.CF10; Core Flood and Decay Heat Train 2 Injection Lines to Reactor Vessel; 17-VT-300
- B15.000.CF09; Core Flood and Decay Heat Train 1 Injection Lines to Reactor Vessel; 17-VT-301
- B15.000.DH16; Pressurizer Spray Line; 17-VT-302

Work Orders:
- 200353901; SGR-RC2A – Replace Power Operated Relief Valve ECP 10-0309-001
- 200380790; PM 5078 RC13A Replace Pressurizer Relief Valve
- 200380791; ECP 09-0116-002 PM 5079 RC13B Replace Pressurizer Relief Valve

- 200428009; AFP 1 Response Time Test
- 200429509; Containment ILRT
- 200433297; ECP 10-0470-006 SGR – Install New Reactor Replacement Vessel Closure Head and Control Rod Drive Mechanism's Post Maintenance (Modification) Testing
- 200464590; SGR – Replace Power Operated Relief Valve (Electrical) ECP 10-0309-001

Other:
- ISTP3; Third Ten Year Inservice Testing Program; Revision 11
- ASME Operation & Maintenance Code, 1995 Edition, 1996 Addenda
- ISTB1; Pump and Valve Basis Document, Volume I – Valve Basis; Revision 10
- ISTB2; Pump and Valve Basis Document, Volume II – Pump Basis; Revision 12
- ISTB3; Pump and Valve Basis Document, Volume III – Stroke Time Basis; Revision 41

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u>

Condition Reports:
- 2011-02491; Unsatisfactory radiological Condition in the Aux Building 565' Elevation
- 2011-02797; Annulus Scaffolding Sealands are Inadequate for Radioactive Storage or Shipment
- 2011-02868; WSI Worker Receives Dose Rate Alarm after Coming to Contact with Piping in No. 2 MPR Pipe Chase
- 2011-03034; Potential Tritium Leak – Condenser Pit Outage Temporary Line to Settling Basin
- 2011-03200; Insulators Receives Dose Rate Alarm While Working in the East "D" Ring 565'
- 2011-03256; Bad Radiological Practice of Men in the Batwing Box with Contaminated Material such as Tool without Wearing Protective Clothing
- 2011-03270; Individual Discovered in Containment with a Rag on his Head Instead of Hood
- 2011-03401; Recent Trending Indicates Improvement is Warranted in Radworker Work Behavior in the Area of Radiological Anti-Contamination Clothing Control
- 2011-03404; Condenser Pit Sump Discharge Leak to Ground
- 2011-03429; Individual did not Meet RWP Requirements
- 2011-03459; NOP-OP-4601 Contamination Control Program Guidance was not Completely Implemented
- 2011-03533; Worker Lost Dosimeter in RCA
- 2011-03535; Contamination Boundary Violation
- 2011-03680; There were an Increased of Radworkers Coaching to use Low Dose Waiting Areas
- 2011-03687; Worker Enters Radiologically Controlled Area without Being on Proper RWP
- 2011-03808; Personnel Contamination Event 17M
- 2011-03808; Personnel Contamination Event 17M-11
- 2011-03847; Radworkers Using Electronic Alarming Dosimeters as Dose Rate Meters
- 2011-03918; Observations Indicate Additional Action is Warranted in the Area of Hoses and Cords not Being Secured at Contaminated area Boundaries
- 2011-03980; Radiological Posting Standards Not Met
- 2011-04057; Method Used to Control VHRA Access to Fuel Transfer Tube from Core Flood Tank 1-1 Area
- 2011-04083; Revise Station Vent Radiation Element Radiation Alarm Setpoints
- 2011-91281-001; Plant Health Committee to Design Engineering to Develop a Plan to Enlarge the Current 3" Diameter Piping from the Condenser Flood Pump to the Settling Basin

Procedures:
- DB-0204-0; Source Leak Test Records

- NOP-OP-4102; Radiological Posting, Labeling and Markings; Revision 7
- NOP-OP-4104; Job Coverage; Revision 0
- NOP-OP-4106; Control of Radiography Operation; Revision 2
- NOP-OP-4107; Radiation Work Permit; Revision 7
- NOP-OP-4204; Special External Exposure Monitoring; revision 5
- NOP-OP-4301; Respiratory Protection Program; Revision 2
- NOP-OP-4702; Air Sampling; Revision 2

Other:
- Davis Besse Methods for Measuring Effective Dose Equivalent from External Exposure for OTSG Work and Reactor Head Repair Activities
- Davis Besse Nuclear Power Station; 17 Mid-Cycle and 17 RFO In-Processing Focus Areas
- 17M Outage Daily Exposure Summary; October 18, 2011
- 17 Mid-cycle Radiation Protection Trending; October 20, 2011
- Bechtel Radiography Management Plan, Davis-Besse Containment Vessel, Fall 2011; Revision 0

2RS2 Occupational ALARA Planning and Controls

Condition Reports:
- 2010-72971; Alloy 600 Dose Delta
- 2010-73156; Elevated Dose Rates on Core Flood Shielded Work Platforms with the Incores Pulled
- 2010-74240; A Review of Alloy-600 Weld Overlays

Procedures:
- DB-HP-01801; ALARA Design Review; Revision 3
- DB-HP-01802; Control of Shielding; Revision 8
- DB-HP-04027; Installed Shielding Inspection and Engineering Evaluation; Revision 4
- DBBP-RP-0018; Guidance for Work In Progress (WIP) ALARA Reviews; Revision 0
- NOP-OP-417-12; ALARA Work In Progress Review; Revision 0
- NOP-OP-4005; ALARA Program; Revision 1
- NOP-WM-7002; Operational ALARA Program; Revision 1

Radiation Work Permits:
- 2010-5016; ALARA Plan: Work In Progress Review; Insulation Work Activities in the Containment; April 17, 2010
- 2010-5018; ALARA Plan: Reactor Canal Decontamination to Include ALARA Plan and associated TEDE ALARA Evaluations for RWP 10010886; Drywell Insulation Activities; Revision 1
- 2010-5104; ALARA Plan: Reactor Head Disassembly/Reassembly Work Activities; January 17, 2010
- 2010-5302; ALARA Plan: Work In Progress Review; OT Steam Generator Platform Work; March 13, 2010
- 2010-5405; All Task: Letdown Cooler Project; ALARA Work In Progress Review; April 21, 2010
- 2010-5600; ALARA Plan: ALARA Work In Progress; Weld Overlays of RCP Cold Leg (4) Suction and (4) Discharge Lines and all Support Activities such as Scaffolding, Insulation, Shielding, UT/PT, and Interference Removal; April 12, 2010
- 2010-5601; ALARA Plan: Alloy-600, Cutting to Access North and South Core Flood Nozzles, Install and Remove shielded Work Platforms; April 30, 2010

- 2010-5602; ALARA Plan; Weld Overlays of North/South Core Flood Nozzles and All Support Activities; February 17, 2010
- 2010-5603; ALARA Plan: Concrete Cutting through Bio-shield to Access North and South Core Nozzles; January 6, 2010
- 2010-5604; ALARA Plan: ALARA Work-In-Progress Review; March 14, 2010

Other:
- ALARA Work In Progress Reviews for RWP 1010-5600; Weld Overlay on Reactor Coolant Pumps; March 16, 2010
- Alloy-600-1R16; Preliminary Estimate Summary: RCP/Drain Line Overlays; Core Flood Nozzle Overlays; and Common Support Activities; April 15, 2010

2RS5 Radiation Monitoring Instrumentation

Condition Reports:
- CR-2011-02061; Flow Rate Information Could Effect Information Reported in the Annual Radiological Environmental Operating Report; September 16, 2011
- CR-11-89820; 4598AA Flow was Recorded Less than Three SCFM; February 18, 2011
- CR-11-87764; RE-4598BA Sample Pump Failed and Caused A Momentary Spurious Spike in Radiation; January 5, 2011
- CR-10-81010; RE-4598BA RIC Flowrate Indication does not Correspond with Local Flowrate Indication; August 10, 2010
- CR-10-83234; Broken Gasket was discovered on the Inner Bottom of the Charcoal Holder; September 27, 2010
- CR-10-81607; Radiological Air Sampler Documentation Deficiencies; August 23, 2010
- CR-11-94845; FT-5090 Appears to be Degraded Due to Both F885 and RIC 4598AA Reading a Spiked Indication of 145 KCFM as Compared 4598BA Which Reads a Steady 125 KCFM; May 15, 2011
- CR-10-79761; Station Vent Accident Range Radiation Appears to Have Gone to the Default Setting; July 16, 2010

Procedures:
- ED-7191-2; Toledo Edison Calculation Sheet; RE-4598AA and 4598BA Setpoint Basis; January 23, 1995
- DB-CN-03008; Station Vent Releases, Weekly Radiological Monitoring, Sampling and Analysis of RE-4598AA; Revision 10
- RA-EP-02240; Davis Besse Emergency Plan Implementing Procedure; Revision 5
- DB-MI-03414; Calibration of Flow Power Supplies and Battery Checks for RE-4597AA, RE-4597BA, RE-4598AA, and RE-4598BA Normal Range Radiation Monitors; Revision 9
- DB-OP-03007; Miscellaneous Instrument Daily Checks; Revision 19
- M-340DQ-00196-02; Instrument Manual for Hastings Linear Gas Flow Probe; Manual
- DB-OP-03011; Radioactive Liquid Batch Release; Revision 19
- DB-HP-01457; Radiation Protection Instrumentation Procedure; MGP-AMP-50/100/200 Calibration and Use; Revision 2
- DB-HP-01456; Radiation Protection Instrumentation Procedure; Calibration of Eberline AMS-4; Revision 2
- DB-HP-01460; Radiation Protection Instrumentation Procedure; RADeco Model H-810 DC Calibration and Use; Revision 0
- DB-HP-01461; Radiation Protection Instrument Procedure; PCM12 Calibration, Source Check and Use; Revision 1
- DB-HP-01455; Radiation Protection Procedure; Operation of Eberline AMS-4; Revision 1

- DB-HP-01453; Radiation Protection Instrumentation Procedure; Continuous Particulate Air Monitor AMS-3, Calibration and Use; Revision 6
- DB-HP-01456; Radiation Protection Instrumentation Procedure; Air Sampler Calibration; Revision 7
- DB-HP-01452; Radiation Protection Procedure; Small Article Monitor Calibration; Revision 3
- DB-HP-01442; Radiation Protection Instrumentation Procedure; MGP Telepole Calibration and Use; Revision 3
- DB-HP-01439; Radiation Protection Procedure; Bicron Labtech; Revision 1
- DB-HP-01438; Radiation Protection Instrumentation Procedure; Frisker Calibration; Revision 5
- DB-HP-01435; Radiation Protection Instrumentation Procedure; Calibration and Use of the Portal Monitor SPM 904C/SPM 906; Revision 3
- DB-HP-01436; Radiation Protection Procedure; DMC 90/100/2000 Calibration and Use; Revision 3
- DB-CN-03005; Radiological Monitoring Weekly, Semi-Monthly and Monthly Sampling; Revision 3
- DB-HP-01432; Radiation Protection Instrumentation Procedure; ASP-1 Calibration and Use; Revision 3
- MS-C-09-10-01; Fleet Oversight Audit Report

Other:
- Davis-Besse Offsite Dose Calculation Manual; Revision 25
- Radiation Monitor Setpoint Manual; RE-4598AAC Vent Normal Range; August 3, 2011

4OA1 Performance Indicator Verification

Forms:
- NOBP-LP-4012-52; Reactor Coolant System Specific Activity; Completed Forms for October 2010 through September 2011
- NOBP-LP-4012-53; Reactor Coolant System Leakage; Completed Forms for October 2010 through September 2011

Procedures:
- DB-CH-06002; Sampling System Nuclear Areas; Revision 28
- NOBP-LP-4012; NRC Performance Indicators; Revision 3
- NOBP-LP-4012; NRC Performance Indicators; dated April 21, 2008
- NOBP-LP-4012-52; Reactor Coolant System Specific Activity Occurrence; Revision 0; from August 2010 through August 2011

Other:
- NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 6
- Select Operator Logs covering the period of October 2010 through September 2011
- Maintenance Rule Unavailability Database covering the period of October 2010 through September 2011

4OA2 Identification and Resolution of Problems

Condition Reports:
- 2011-03346: Fractured Concrete Found at 17M Shield Building Construction Opening
- 2011-03996; Extent of Condition for Shield Building Fracture Indications
- 2011-04190; Surface Cracks Identified on Fluted Areas of the Shield Building
- 2011-04214; Core Bore Found Additional Crack in Architectural Flute Area

- 2011-04402; Fractured Concrete Found at 17M Shield Building at Main Steam Line Penetrations
- 2011-04507; Isolated Crack Indication Identified by Impulse Response Testing
- 2011-04648; Shield Building IR Indications above Elevation 780
- 2011-05475; Concrete Cracking at the Top of the Shield Building Wall
- 2011-05648; Concrete Cracking in Shoulder 4 / Flute 2 Region of the Shield Building (Azimuth 67.5)
- 2011-05726; Unable to Perform Section 4.14, Pressurizer Liquid Space Sampling During Pressurizer Heatup, of DB-CH-06002, Sampling System Nuclear Areas, as Written
- 2011-05735; Final Portion of Shield Building Restoration Included a Cement Not Tested By a Safety-Related (Q) Laboratory
- 2011-05777; Void in Concrete at the top of the Shield Building pour back (Bechtel NCR 20)
- 2011-05781; PI1507A For No. 1 Decay Heat Pump Over-Ranged
- 2011-05782; PI1538A For Decay Heat Pump 2 Was Over-Ranged
- 2011-05794; Bechtel Out-of-Process for Authorizing Use-as-Is on Shield Building Opening Restoration Nonconformance
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06185; Error in Calculation C-CSS-099.20-056, Revision 01

Calculations:
- C-CSS-099.20-046; Evaluation of Shield Building for the Permanent Condition; Revision 0
- C-CSS-099.20-047; Restoration of Shield Building Construction Opening; Revision 0
- C-CSS-099.20-053; Evaluation of Shield Building for the Interim Condition with Outside Vertical Reinforcement Removed at Each Flute Shoulder; Revision 0
- C-CSS-099.20-054; Evaluation of Shield Building for the Permanent condition with Outside Vertical Reinforcement Removed at Cracking Areas; Revisions 0, 1, 2, and 3
- C-CSS-099.20-055; II/I Evaluation for Architectural Flute Shoulder; Revision 0
- C-CSS-099.20-056; Evaluation of Shield Building Hoop Reinforcement with Observed Cracking; Revisions 0 and 1

Drawings:
- C-111A; Shield Building Exterior Developed Elevation; Revision 0 and 1

Procedures:
- NOP-LP-2001; Corrective Action Program; Revision 29
- NOBP-LP-2010; FENOC Trend Coding; Revision 10
- DB-CH-06002; Sampling System Nuclear Areas; Revision 28
- DB-PF-03811; Miscellaneous Valves Test; Revision 19
- DB-OP-06904; Shutdown Operations; Revision 38

Other:
- FENOC Quality Assurance Program Manual; Revision 15
- Davis-Besse Shield Building Investigation and Technical Summary; Revisions 0 and 1
- Davis-Besse Shield Building Cracking Investigation and Assessment Report; Revisions 0 and 1

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- 2006-00624; Water Spray on Motor Control Centers E11B and E11C

- 2007-32157; Water Spraying Out of the Overhead Between No. 3 and No. 4 Mechanical Penetration Rooms
- 2008-45463; Water Dripping on E11C
- 2010-87048; RC13A and RC13B Fail As-Found Testing at Vendor
- 2011-05456; E11C Water Intrusion / Fire
- 2011-05457; 4-Way Ringdown Not Ringing at State and Lucas County During Alert Classification
- 2011-05465; Emergency Response Facility Computer Issues During Alert
- 2011-05466; Recovery From the 11/16/2011 Davis-Besse Alert
- 2011-05523; Second Control Power Transformer Found Damaged as a Result of MCC E11C Fire

Procedures:
- RA-EP-01500; Emergency Classification; Revision 14
- DB-OP-02529; Fire Procedure; Revision 5
- DB-OP-02501; Serious Station Fire; Revision 16
- DB-OP-06241; Auxiliary Boiler Operating Procedure; Revision 24
- DB-OP-00000; Conduct of Operations; Revision 19
- NOP-OP-01014; Plant Status Control; Revision 1
- NG-QS-00121: Davis-Besse Procedure Writer’s Guide; Revision 5

4OA5 Other Activities

Condition Reports:
- 2011-03346: Fractured Concrete Found at 17M Shield Building Construction Opening
- 2011-03996; Extent of Condition for Shield Building Fracture Indications
- 2011-04190; Surface Cracks Identified on Fluted Areas of the Shield Building
- 2011-04214; Core Bore Found Additional Crack in Architectural Flute Area
- 2011-04402; Fractured Concrete Found at 17M Shield Building at Main Steam Line Penetrations
- 2011-04507; Isolated Crack Indication Identified by Impulse Response Testing
- 2011-04648; Shield Building IR Indications above Elevation 780
- 2011-05475; Concrete Cracking at the Top of the Shield Building Wall
- 2011-05648; Concrete Cracking in Shoulder 4 / Flute 2 Region of the Shield Building (Azimuth 67.5)
- 2011-05735; Final Portion of Shield Building Restoration Included a Cement Not Tested By a Safety-Related (Q) Laboratory
- 2011-05770; Bechtel Subcontractor Certifications not Transmitted to FENOC for Review
- 2011-05777; Void in Concrete at the top of the Shield Building pour back (Bechtel NCR 20)
- 2011-05794; Bechtel Out-of-Process for Authorizing Use-as-Is on Shield Building Opening Restoration Nonconformance
- 2011-05795; Concrete Void at Top of Shield Building Restoration Larger than Previously Reported per CR 2011-05777
- 2011-05804; Conditional Release of Concrete for Shield Building Void
- 2011-05904; Errors Identified in Shield Building Crack Calculation C-CSS-059.20-056
- 2011-06185; Error in Calculation C-CSS-099.20-056, Revision 01
- 2011-04373; NRC Potential Violation Regarding RRVCH Stud Hole Exams; dated October 26, 2011
- 2011-03772; Visual Inspection of Containment Vessel Attachment Welds not Performed; dated October 15, 2011
- 2011-03771; Missed MT on Temporary Weld; dated October 15, 2011

- 2011-01739; NRC Surface Examination of Accessible Internal Surface of RRVCH Stud Holes; dated September 9, 2011
- 2011-00344; No Surface Exam of the RRVCH Stud Holes; dated August 9, 2011

Calculations:
- C-CSS-099.20-046; Evaluation of Shield Building for the Permanent Condition; Revision 0
- C-CSS-099.20-047; Restoration of Shield Building Construction Opening; Revision 0
- C-CSS-099.20-053; Evaluation of Shield Building for the Interim Condition with Outside Vertical Reinforcement Removed at Each Flute Shoulder; Revision 0
- C-CSS-099.20-054; Evaluation of Shield Building for the Permanent condition with Outside Vertical Reinforcement Removed at Cracking Areas; Revisions 0, 1, 2, and 3
- C-CSS-099.20-055; II/I Evaluation for Architectural Flute Shoulder; Revision 0
- C-CSS-099.20-056; Evaluation of Shield Building Hoop Reinforcement with Observed Cracking; Revisions 0 and 1

Drawings:
- C-111A; Shield Building Exterior Developed Elevation; Revision 0 and 1
- Bechtel drawing; 000-DB-1000-000001; Shield Building Temporary Construction Opening Concrete Preparation; Revision 2
- Bechtel drawing; 25539-200-C0K-1002-00048; Layer 2 Horizontals (Outside Rebar Mat); Revision 0

Other:
- Davis-Besse Shield Building Investigation and Technical Summary; Revisions 0 and 1
- Davis-Besse Shield Building Cracking Investigation and Assessment Report; Revisions 0 and 1
- AMEC Report of Concrete Mixer Uniformity Testing; dated October 12, 2011
- Bechtel Nonconformance Report 25539-200-G61-GCE-00002; dated October 13, 2011
- 25539-000-3PS-DB02-Q00001; Concrete Work for Safety-Related Applications; Revision 1
- 25539-000-3PS-DB01-Q00001; Purchase of Ready-Mix Concrete for Safety-Related Applications; Revision 5
- 25539-000-3PS-SY01-Q0001; Material Testing Services; Revision 5
- AMEC Concrete Field and Lab Test Report; dated November 2, 2011
- Production and Sister Splice Report; dated November 17, 2011
- Reinforcing Bar Mechanical Splicing Qualification Form; dated October 7, 2011
- Interim Report Camtack and Bargrip Sleeve Testing for Dayton Barsplice Inc.; dated September 18, 1979
- BASF COC Letter Pozzolith 200N; Lot 2004-78841V11; dated July 19, 2011
- BASF COC Letter Rheobuild 1000; Lot 2004-67107011; dated July 19, 2011
- BASF COC Letter Air Entraining Admixture for Concrete; Lot 2004-77071U11; dated July 19, 2011
- ECP 10-0458-002; SGR-17M- Restore Shield Building Wall at Construction Opening; Revision 0
- Procedure Demonstration Qualification Summary No. 449, Procedure 54-ISI-801-00" UT of PWR Shell Welds"; Revision 0
- 25539-200-V1A-FU00-00001-001; Type 3 Dedication Plan - No. 8 to No. 8 Bargrip XL Coupler, Barsplice Part No. 8XL
- 25539-200-V1A-FU00-00001-001; Type 3 Dedication Plan - No. 10 to No. 8 Bargrip XL Coupler, Barsplice Part No. 10/8XL
- 25539-200-V1A-FU00-00001-001; Type 3 Dedication Plan - No. 11 to No. 11 Bargrip XL Coupler, Barsplice Part No. 11XL

- 25539-200-V1A-FU00-00001-001; Type 3 Dedication Plan - No. 10 to No. 10 Bargrip XL Coupler, Barsplice Part No. 10XL
- Concrete Field and Lab Test Report 002; dated November 2, 2011
- Concrete Field and Lab Test Report 003; dated November 2, 2011
- Concrete Field and Lab Test Report 006; dated November 22, 2011
- Concrete Field and Lab Test Report 009; dated November 22, 2011
- Concrete Field and Lab Test Report 010; dated November 22, 2011
- Rebar Splice Test Inspection Record; Splice Number V-1; ID Number 190
- Rebar Splice Test Inspection Record; Splice Number V-2; ID Number 190
- Rebar Splice Test Inspection Record; Splice Number H-1; ID Number 190
- Rebar Splice Test Inspection Record; Splice Number H-2; ID Number 190
- Rebar Splice Test Inspection Record; Splice Number V-1; ID Number 204
- Rebar Splice Test Inspection Record; Splice Number V-2; ID Number 204
- Rebar Splice Test Inspection Record; Splice Number H-1; ID Number 204
- Rebar Splice Test Inspection Record; Splice Number H-2; ID Number 204
- Rebar Splice Test Inspection Record; Splice Number V-1; ID Number 114
- Rebar Splice Test Inspection Record; Splice Number V-2; ID Number 114
- Rebar Splice Test Inspection Record; Splice Number H-1; ID Number 114
- Rebar Splice Test Inspection Record; Splice Number H-2; ID Number 114
- SKZ904; Shield Building Exterior Developed Elevation; Revision 0
- SM02; Swaging Instructions; BPI-Grip Couplers; March 2010
- SP-701; Consolidated Power Supply, Dedication of Commercial Grade Items; Revision 14
- QC Record of Weld Heat Input - Door Sheet FW-1 Weld; dated November 2, 2011

Radiographic Records:
- Computed Radiographic Image Interpretation Sheets FW-1; dated November 11, 2011
- Computed Radiographic Image Interpretation Sheets FW-1 Repairs; dated November 12, 2011
- Computed Radiographic Technique Report and Evaluation Sheets FW-1; dated November 10, 2011
- Computed Radiographic Technique Report and Evaluation Sheets FW-1 Repairs; dated November 12, 2011

Non Destructive Examination Records:
- AC Magnetic Particle Examination Data Sheet Report MT-02; RRVCH Stud Holes; dated October 1, 2011
- AC Magnetic Particle Examination Data Sheet Report MT-02; RRVCH Stud Holes; dated November 1, 2011
- Magnetic Particle Examination Report MT-043; CV Plate Door Sheet Weld FW-1 Annulus Side; dated November 7, 2011
- Magnetic Particle Examination Report MT-040; CV Plate Door Sheet Weld FW-1 Containment Side; dated November 5, 2011

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ARM	Area Radiation Monitor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BACC	Boric Acid Corrosion Control
CAM	Continuous Air Monitor
CAP	Corrective Action Program
CC	Construction Code
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
CV	Containment Vessel
DC	Direct Current
deg F	Degrees Fahrenheit
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
EMI/RFI	Electromagnetic Interference/Radio Frequency Interference
ET	Eddy Current
FRV	Feedwater Regulating Valve
FW	Feedwater
HPI	High Pressure Injection
I&C	Instrumentation and Controls
ICS	Integrated Control System
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report *or* Impulse Response
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MOV	Motor-Operated Valve
MT	Magnetic Particle
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
ODCM	Offsite Dose Calculation Manual
OpESS	Operating Experience Smart Sample
PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance

PMT	Post-Maintenance Testing
psig	Pounds Per Square Inch Gauge
PT	Dye Penetrant
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RFO	Refueling Outage
RG	Regulatory Guide
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RRVCH	Replacement Reactor Vessel Closure Head
RT	Radiographic
RWP	Radiation Work Permit
SB	Shield Building
SDP	Significance Determination Process
SG	Steam Generator
SL	Severity Level
SMAW	Shielded Metal Arc Weld
SRO	Senior Reactor Operator
SSC	Structures, Systems and Components
SW	Service Water
TIA	Task Interface Agreement
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WO	Work Order

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-346
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Enclosure: Inspection Report 05000346/2011005
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Letter to B. Allen from J. Cameron dated January 31, 2012.

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION INTEGRATED INSPECTION REPORT 05000346/2011005

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