UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

August 24, 2011

ML112360693

EA-11-142

Joseph Kowalewski
Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC RADIATION SAFETY INSPECTION REPORT 05000382/2011009; PRELIMINARY WHITE FINDING

Dear Mr. Kowalewski:

This letter discusses a finding that has preliminarily been determined to be White, a finding with low to moderate increased safety significance that may require additional NRC inspections. As described in the enclosed report, Waterford Steam Electric Station, Unit 3, failed to use effective engineering controls as part of pre-job planning to reduce contamination and subsequent exposure. Waterford Steam Electric Station, Unit 3, failed to keep highly radioactive water from leaking onto the work areas around the reactor coolant pumps, despite having knowledge that this condition could occur. This failure resulted in high levels of radioactive contamination and unexpected and unintended radiation dose to plant workers. The finding was assessed based on the best available information, using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The finding was preliminarily determined to be White because (1) this was an ALARA planning issue, (2) the site's three-year average collective dose exceeded 135 person-rem and (3) one of the work activities accrued more than 25 person-rem or alternately, the finding would still be preliminarily determined to be White because there were more than four other occurrences in which the actual collective dose exceeded 5 person-rem, and the estimated/planned dose by more than 50 percent. The final resolution of this finding will be conveyed in separate correspondence.

As described in Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," a finding may or may not be associated with regulatory non-compliance and, therefore, may or may not result in a violation. Based on the review of this issue and in accordance with Inspection Manual Chapter 0612, the NRC determined that no violation of a regulatory requirement occurred. The Waterford Steam Electric Station, Unit 3, ALARA program was adequate, in most respects, and a violation of 10 CFR 20.1101(b) did not exist, because there was only one performance deficiency, whereas an ALARA violation is normally associated with multiple ALARA performance deficiencies and/or numerous causes. Additionally, the inspectors did not identify an inadequacy associated with radiation protection procedures required by Technical Specification 6.8.1 and Regulatory Guide 1.33.

In accordance with Inspection Manual Chapter 0609, "Significance Determination Process," we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final significance determination process determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of Inspection Manual Chapter 0609.

Please contact Mr. Gregory E. Werner at (817) 860-8156 within 10 business days of the date of receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

No violation of regulatory requirements occurred. However, since the NRC has not made a final determination in this matter, please be advised that the number and characterization of findings described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agency-wide Document Access Management System (ADAMS). ADAMS is accessible from the NRC Website at www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Anton Vegel, Director
Division of Reactor Safety

Docket No.: 50-382
License No.: NPF-38

Enclosure:
NRC Inspection Report 05000382/2011009
w/Attachments:
1. Supplemental Information
2. White Paper – Preliminary Communication
3. White Paper – Supplemental Communication
4. Response to Supplemental Questions
5. Waterford 3 Response to Supplemental Questions
6. Waterford 3 RCP Leakage Flow Experience
7. Untitled Clarification of Information in White paper

cc w/enclosure:
Distribution via Listserv for Waterford Steam Electric Station, Unit 3

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R:\ ADAMS ML

ADAMS: ☐ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: LTR
	☒ Publicly Available	☒ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

HP:PSB2	SHP:PSB2	C:PSB2	C:DRPE	SHP:NRR/DIRS/IHBP
NGreene	LRicketson	GWerner	JDrake	SGarry
/RA/	*/RA/*	*/RA/*	*/RA/*	*/RA/ E-mail*
8/11/2011	8/11/2011	8/16/2011	8/16/2011	8/17/2011
ES:OE/EB	SES:ACES	DD:DRS	D:DRS	
NColeman	RKellar	TBlount	AVegel	
/RA/ per e-mail	*/RA/*	*/RA/*	*/RA/*	
8/22/2011	*8/19/2011*	8/22/2011	8/23/2011	

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

Enclosure:
NRC Inspection Report 05000382/2011009
w/Attachments:

1. Supplemental Information
2. White Paper – Preliminary Communication
3. White Paper – Supplemental Communication
4. Response to Supplemental Questions
5. Waterford 3 Response to Supplemental Questions
6. Waterford 3 RCP Leakage Flow Experience
7. Untitled Clarification of Information in White paper

cc w/enclosure:
Distribution via Listserv Waterford Steam Electric Station, Unit 3

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000382
License	NPF-38
Report:	05000382/2011009
Licensee	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location	17265 River Road Killona, LA 70057-0751
Dates:	March 14 through August 10, 2011
Inspectors:	L. Ricketson, PE, Senior Health Physicist N. Greene, PhD, Health Physicist
Approved By:	Gregory E. Werner, Chief Plant Support Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000382/2011009; 3/14/2011 – 8/10/2011; Waterford Steam Electric Station, Unit 3, Regional Report; Occupational ALARA Planning and Controls

The report covered approximately a 5-month period of inspection by two region-based inspectors. One White finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The crosscutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level 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 Findings and Self-Revealing Findings

Cornerstone: Occupational Radiation Safety

- TBD. The inspectors identified an apparent White finding because the licensee failed to use effective engineering controls as part of pre-job planning to reduce contamination and subsequent exposure. The primary reason for the dose overage was the licensee's failure to prevent radioactive water from leaking into work areas and raising radiation dose rates. As corrective action, the licensee installed a trough system to collect and route the radioactive water away from the work area and to the reactor containment floor drain system. This issue was placed in the corrective action program as Condition Report CR-WF3-2011-05672.

  The failure to use effective engineering controls as part of pre-job planning to reduce contamination and subsequent exposure is a performance deficiency. The finding is more than minor because it was similar to (the more than minor) Example 6.i in Inspection Manual Chapter 0612, Appendix E, "Example of Minor Issues," in that the actual collective dose exceeded 5 person-rem and exceeded the planned, intended dose by more than 50 percent. Additionally, the finding is associated with the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective in that it increased collective radiation dose. The inspectors used Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," to analyze the significance of the finding. The finding was preliminarily determined to be White (low to moderate safety significance) because it involved ALARA planning or work controls; the average collective dose at the time the finding was identified was greater than 135 person-rem; and the actual dose associated with a work activity was greater than 25 person-rem. Alternately, there were greater than four occurrences in which the actual collective dose exceeded 5 person-rem and the estimated/planned dose by more than 50 percent. The final significance of this finding is to be determined. The finding had a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component, because the licensee did not institutionalize operating experience concerning the effects of reactor coolant pump leakage on work area dose rates [P2.(b)] (Section 2RS02).

B. **Licensee-Identified Violations**

None

REPORT DETAILS

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

During an inspection conducted November 15 - 18, 2010, the inspectors opened two unresolved items involving issues of concern which may have impacted the collective dose accrued during the 2009 refueling outage. An unresolved item is an issue of concern about which more information is required to determine (a) if a performance deficiency exists, (b) if the performance deficiency is more than minor, or (c) if the issue of concern constitutes a violation. Unresolved Item 05000382/2010005-02 involved removal of radioactivity from the steam generators. Unresolved Item 05000382/2010005-03 involved leakage from the reactor coolant pump seals. The licensee provided additional information necessary for resolution through a series of submissions. The inspectors performed an in-office review of the information. The licensee's responses which provided the additional information are included as Attachments 2 - 7 to this report. Other documents reviewed during this inspection are listed in Attachment 1.

b. Findings

(Closed) Unresolved Item 05000382/2010005-03; Leakage from the Reactor Coolant Pump Seals

Introduction: The inspectors identified a preliminary White finding for the failure to use effective engineering controls as part of pre-job planning to reduce contamination and subsequent exposure.

Description: The licensee experienced high personnel doses during the 2009 refueling outage and had a high three-year average collective dose for a pressurized water reactor. Refueling Outage 16 was conducted October 19 through December 4, 2009. The licensee's outage dose goal was 125 person-rem. The actual dose, measured by electronic dosimeters, was nearly 273 person-rem. The licensee's 2007-2009 3-year, collective dose average was 136.478 person-rem. The inspectors reviewed the post-job reviews of 13 work activities that accrued greater than 5 person-rem. Twelve of the work activities exceeded their originally planned dose estimates by more than 50 percent. After reviewing the causes listed by the licensee in the post-job reviews and interviewing licensee personnel, the inspectors determined that for five work activities the licensee failed to use adequate engineering controls to prevent the spread of radioactive water to work areas which resulted in high dose rates.

The licensee had a history of leaking reactor coolant pump seals. Radioactive water from the reactor coolant system leaked from the reactor coolant pump vapor seals and onto the reactor coolant pumps and nearby work areas, raising the dose rates. The apparent cause evaluation associated with Condition Report CR-WF3-2009-7262, stated in part:

> During reactor coolant system depressurization to reach mode 5 and at approximately 350 psia in the reactor coolant system, as anticipated, all four reactor coolant pump seal vapor stages de-staged and leaked highly contaminated reactor coolant system water onto pump insulation, adjacent structures, and to -11 [foot elevation of the] reactor containment building. As a result of this leakage, dose rates in the immediate vicinity of the reactor coolant pumps were elevated to 3.5 rem/hour. Contamination levels were up to 500 mrad/hour.

According to Condition Report CR-WF3-2005-3831, the licensee has known of the reactor coolant pump vapor seals de-staging and leaking reactor coolant water outside of the reactor coolant pump seal bowl area since 2005, when it was identified during a containment entry after shutdown. This leakage was a result of a reactor coolant seal design change installed during Refueling Outage 12, in the Fall of 2003. Since 2005, at least one reactor coolant pump vapor seal has de-staged and leaked radioactive water each time the plant has shutdown and depressurized. The inspectors asked licensee representatives if the reactor coolant leaked onto reactor coolant pump insulation and into work areas each time the reactor coolant pump seals leaked. The licensee acknowledged it had, but stated the leakage had less impact previously on dose than during Refueling Outage 16. Although, during Refueling Outage 14 (November 2006), reactor coolant pumps 1B and 2B leaked onto the respective reactor coolant pump insulation packages and raised the effective dose rate for reactor coolant pump work as much as 35 percent. The doses were higher again during Refueling Outage 16 because the licensee manually tripped the plant from approximately 100 percent reactor power. According to Condition Report CR-WF3-2009-07262, this caused a thermal hydraulic shock which resulted in a release of activated corrosion and wear products (crud).

The effect of the leakage (higher collective dose) could have been reduced had the radioactive water been controlled and not allowed to leak onto reactor coolant pump insulation and other work areas. The licensee discovered a design problem with the reactor coolant pump vapor seal leak-off lines' ability to handle radioactive water leakage as early as 2003. The apparent cause evaluation associated with Condition Report CR-WF3-2009-5501 stated, in part, "The vapor stage seal leak-off line was not performing its design function." This statement was listed as a "Lesson-Learned" associated with the following condition reports: CR-WF3-2003-3006, CR-WF3-2003-3692, CR-WF3-2005-1361, CR-WF3-2005-3831, CR-WF3-2005-3867, CR-WF3-2006-3594, CR-WF3-2006-3597, CR-WF3-2006-3600, CR-WF3-2006-3602, CR-WF3-2007-3536, CR-WF3-2007-3659, and CR-WF3-2008-2659, indicating the licensee has had knowledge of the problem with the performance of the leak-off lines for years and had not effectively addressed the problem. In December 2009, the licensee installed a stainless steel trough around each reactor coolant pump seal housing as part of Engineering Change 18520. These troughs routed the vapor seal leakage to the reactor containment building floor drain. Inspectors noted this modification prevented the radioactive water from leaking onto reactor coolant pump insulation and surrounding equipment during Refueling Outage 17 (spring 2011), thereby reducing exposure rates in the work area.

The licensee provided reasons why the actual collective dose of the five work activities of interest exceeded the estimated/planned dose. The licensee's explanation for each work activity that was affected by reactor coolant pump seal leakage is listed below with the inspectors' evaluations and conclusions.

Radiation Work Permit 20090513, "Reactor Coolant Pump 1A," was estimated to accrue 13.590 person-rem, but the actual dose was 29.692 person-rem. The licensee provided the following reasons the actual dose exceeded the planned dose:

REASON	DOSE (rem)
Discovery of light erosion pattern on rotating baffle	.017
Discovered significant wear on reactor coolant pump 1A motor lower oil cooler return piping	.202
Seismic link field fit-up issues due to unforeseen clearance conditions	.822
Reactor coolant pump 1A main power cable fit-up due to unforeseen manufacturing issue	.242
Discovery of vendor phase rotation issue (motor)	.278
High reactor coolant system activity from hard reactor trip, high dose rates on insulation packages from reactor coolant pump vapor seal de-staging	6.030
Driver mount volute housing separating due to unforeseen thermal binding	2.500
TOTAL	10.091

The licensee accounted for only 10.091 person-rem of the 16.102 person-rem difference and stated that they did not have a justifiable reason for the additional 6.011 person-rem. The inspectors reviewed the licensee's reasons the actual dose exceeded the dose estimate for Radiation Work Permit 20090513 and concluded all the reasons represented unforeseeable problems except for the "high reactor coolant system activity from hard reactor trip, high dose rates on insulation packages from reactor coolant pump vapor seal de-staging." The inspectors determined that the licensee had been aware of the adverse impact on radiation dose from the leaking vapor seals since 2005, which enabled it to foresee this cause and should have resulted in the licensee taking actions to correct this issue, thus minimizing or preventing the additional dose. Therefore, the licensee was given credit for 4.061 person-rem, because this value represented causes which were unforeseeable and was not reasonably within the licensee's ability to prevent, but not for an additional 6.030 person-rem, which represented the additional dose that resulted from the uncontrolled reactor coolant pump leakage. This justifiable dose was added to the original estimate of 13.590 person-rem, for a revised estimate of 17.651 person-rem. The inspectors found the actual dose exceeded the revised estimate by 68 percent.

Radiation Work Permit 20090600, "Health Physics Surveys and Postings in the Reactor Containment Building and Fuel Handling Building," was estimated to accrue 6.923 person-rem, but the actual dose was 15.604 person-rem. The licensee provided the following reasons the actual dose exceeded the planned dose:

REASON	DOSE (rem)
Leaking reactor coolant pump vapor stage seals; crud plate-out in reactor coolant system piping from hard reactor trip (D-rings)	4.528
Extension of outage from planned 32 days to 46 days	2.240
TOTAL	6.768

The licensee accounted for only 6.768 person-rem of the 8.681 person-rem difference, and stated that they did not have a justifiable reason for the additional 1.913 person-rem. The inspectors reviewed the reasons the actual dose exceeded the dose estimate for Radiation Work Permit 20090600 and concluded, as in the previous radiation work permit, the licensee could have foreseen and minimized and/or prevented the radioactive water from the leaking reactor coolant pump seals from contaminating the work area. The licensee combined two sources together (vapor seal leakage and crud plate-out) because it could not determine individual dose contributions. The licensee stated the vapor seal leakage was the major contributor. The inspectors considered the dose contributed by the extended outage duration to be a justified reason for revising the estimated/planned dose and added an additional 2.240 person-rem to the original dose estimate of 6.923 person-rem, for a revised estimate of 9.163 person-rem. The inspectors found the actual dose exceeded the revised estimate by 70 percent.

Radiation Work Permit 20090601, "Reactor Building/Fuel Building Laydown Areas," was estimated to accrue 2.664 person-rem, but the actual dose was 6.271 person-rem. The licensee provided the following reasons the actual dose exceeded the planned dose:

REASON	DOSE (rem)
Leaking reactor coolant pump vapor stage seals; crud plate-out in reactor coolant system piping from hard reactor trip (D-rings)	2.793
TOTAL	2.793

The licensee accounted for only 2.793 person-rem of the 3.607 person-rem difference, and stated that they did not have a justifiable reason for the additional 0.814 person-rem. The inspectors reviewed the licensee's reasons the actual dose exceeded the dose estimate for Radiation Work Permit 20090601 and concluded, as in the previous radiation work permit, the licensee could have foreseen and prevented the radioactive water from the leaking reactor coolant pump seals from contaminating the work area. Therefore, there was no justified reason for increasing the estimated dose. The inspectors found the actual dose exceeded the original estimate by 135 percent.

Radiation Work Permit 20090606, "Entries into the Radiologically Controlled Area and High Radiation Areas for Minor Maintenance," was estimated to accrue 4.647 person-rem, but the actual dose was 10.446 person-rem. The licensee provided the following reasons the actual dose exceeded the planned dose:

REASON	DOSE (rem)
Leaking reactor coolant pump vapor stage seals; crud plate-out in reactor coolant system piping from hard reactor trip (D-rings)	2.013
Extension of outage from planned 35 days to 46 days	1.804
TOTAL	3.817

The licensee accounted for only 3.817 person-rem of the 5.799 person-rem difference, and stated that they did not have a justifiable reason for the additional 1.982 person-rem. The inspectors reviewed the licensee's reasons the actual dose exceeded the dose estimate for Radiation Work Permit 20090606 and concluded, as in the previous radiation work permit, the licensee could have foreseen and prevented the radioactive water from the leaking reactor coolant pump seals from contaminating the work area. The inspectors considered the dose contributed by the extended outage duration to be a justified reason for revising the estimated/planned dose and added an additional 1.804 person-rem to the original dose estimate of 4.647 person-rem, for a revised estimate of 6.451 person-rem. The inspectors found the actual dose exceeded the revised estimate by 62 percent.

Radiation Work Permit 20090610, "Erect and Dismantle Scaffold," was estimated to accrue 9.371 person-rem, but the actual dose was 22.447 person-rem. The licensee provided the following reasons the actual dose exceeded the planned dose:

REASON	DOSE (rem)
Leaking reactor coolant pump vapor stage seals; crud plate-out in reactor coolant system piping from hard reactor trip (D-rings)	5.846
Expanded scope/discovery (unforeseen scaffold installation and adjustments)	4.158
TOTAL	10.004

The licensee accounted for only 10.004 person-rem of the 13.076 person-rem difference, and stated that they did not have a justifiable reason for the additional 3.072 person-rem. The inspectors reviewed the licensee's reasons the actual dose exceeded the dose estimate for Radiation Work Permit 20090610 and concluded, as in the previous radiation work permit, the licensee could have foreseen and prevented the radioactive water from the leaking reactor coolant pump seals from contaminating the work area. The inspectors considered the dose contributed by the expanded scope/discovery to be a justified reason for revising the estimated/planned dose and added an additional 4.158 person-rem to the original dose estimate of 9.371 person-rem, for a revised estimate of 13.529 person-rem. The inspectors found the actual dose exceeded the revised estimate by 66 percent.

Analysis: The licensee's failure to use effective engineering controls as part of pre-job planning to reduce contamination and subsequent exposure was a performance deficiency. The licensee had the ability to foresee and correct or minimize the leaks which resulted in higher collective radiation dose. The finding was more than minor because it was similar to (the more than minor) Example 6.i in Inspection Manual Chapter 0612, Appendix E, "Example of Minor Issues," in that the actual collective dose exceeded 5 person-rem and exceeded the planned, intended dose by more than 50 percent. Additionally, the finding was associated with the program and process attribute of the Occupational Radiation Safety cornerstone and adversely affected the cornerstone objective, in that it increased collective radiation dose.

The inspectors used Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," to analyze the significance of the finding. The finding was preliminarily determined to be White (low to moderate safety significance) because it involved ALARA planning or work controls; the average collective dose at the time the finding was identified was greater than 135 person-rem (136.478 person-rem); and the actual dose associated with a work activity (Radiation Work Permit 20090513) was greater than 25 person-rem (29.692 person-rem). Alternately, there were greater than four occurrences in which the actual collective dose exceeded 5 person-rem, and the estimated/planned dose by more than 50 percent. The finding does not present an immediate safety concern because, after the identification of the finding, the licensee installed a trough system to collect and route the radioactive water away from the work area and to the reactor containment floor drain system. The finding had a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component, because the licensee did not institutionalize operating experience concerning the effects of reactor coolant pump leakage on work area dose rates [P2.(b)].

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. However, the performance deficiency resulted from the failure to meet self-imposed standards. Corporate ALARA administrative control procedure EN-RP-110, "ALARA Program," Revision 6 (not an implementing procedure required by Regulatory Guide 1.33) describes responsibilities for various positions, such as, "Evaluating and recommending use of engineering controls and operational controls to reduce dose, control airborne radioactivity and surface contamination," and

"Reviewing maintenance and modification packages to verify lessons learned and other exposure saving ideas are incorporated into work packages." In addition, administrative Procedure EN-RP-110, "ALARA Program," Revision 6, Section 5.6, in part, states that systems, components, work areas, and procedures are evaluated for compliance per Regulatory Guide 8.8. Regulatory Guide 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 2. Regulatory Guide 8.8, Section C.2.i, addresses leaks from pumps and states, in part, that "Drains on pump housings can reduce the radiation field from this source during servicing. Provision for the collection of such leakage or disposal to a drain sump is appropriate." Specifically, the licensee did not include consideration of engineering controls that were capable of controlling leaks from pump housings to reduce radiation fields during maintenance activities (servicing). Because this finding does not involve a violation but has potentially greater than very low safety significance (to be determined), it is identified as FIN (TBD) 05000382/2011009-01: "Failure To Use Effective Engineering Controls As Part Of Pre-Job Planning To Reduce Contamination And Subsequent Exposure."

4OA5 Other Activities

(Closed) Unresolved Item 05000382/20100005-02, "Removal of Radioactivity from the Steam Generators"

a. Inspection Scope

In NRC Inspection Report 50-382/2010005, the inspectors identified an unresolved item involving the retention of highly radioactive water in the steam generators. The steam generators were not drained in the usual timeframe as in all previous outages. In previous outages, the steam generators were usually drained by the fourth day of the outage. During Refueling Outage 16, the steam generators were not drained until day 17 (after the reactor was completely defueled). Shutdown cooling was initiated on October 22, 2009 (the fourth day of the outage), and resulted in low flow of water through the steam generators and little clean up of the highly radioactive water in the steam generators. This combined with a Cobalt-58 peak of 5.0 microcuries per milliliter allowed settling and further plate out of crud in the steam generators. This was an issue of concern because the actual radioactive concentration of Cobalt-58 was considerably greater than the recommended target value of less than 0.05 microcuries per milliliter for Cobalt-58 as listed in Procedure CE-002-006, "Maintaining Reactor Coolant System Chemistry," Revision 305, and could have contributed to the higher than anticipated work activity doses. The inspectors concluded more information was required related to the licensee's decision to not remove additional radioactivity from the steam generators before it could be determined if a performance deficiency existed. The NRC asked the licensee to provide the required information during a teleconference conducted December 1, 2010. The licensee responded, stating, "Draining the steam generators to remove the highly contaminated water was not considered a reasonable option because that would have required us to install the nozzle dam prior to defueling. Going to reduced inventory/mid-loop to install the nozzle dams is a significant challenge to nuclear safety that we will avoid when possible. During Refueling Outage 16, it was possible to avoid reduced inventory and get the core offloaded without the additional risk of reduced inventory operation. Plant Risk was maintained Green during the entire outage as an outcome...." Flushing the steam generators to remove the highly contaminated water was not considered a reasonable action because this would have required us to continue to operate the reactor coolant pumps for an extended period" [instead of initiating the shutdown purification system]. The inspectors concluded the licensee's actions to not drain down the reactor coolant system with fuel in the vessel and increase risk was appropriate and this did not constitute a performance deficiency.

4OA6 Meetings

Exit Meeting Summary

On August 10, 2011, the inspectors presented the results of the radiation safety inspection to Mr. C. Arnone, General Manager, Plant Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Kowalewski, Site Vice President
C. Arnone, General Manager, Plant Operations
D. Boan, Supervisor, Radiation Protection
J. Brawly, ALARA Supervisor, Radiation Protection
C. England, Manager, Radiation Protection
W. Hardin, Licensing, Specialist
J. Hornsby, Manager, Chemistry
R. Murillo, Acting Director, Nuclear Safety Assurance
C. England, Manager, Radiation Protection
J. Pollack, Engineer, Licensing
J. Ridgel, Quality Assurance Manager
W. Steelman, Manager, Licensing

NRC Personnel

M. Davis, Senior Resident Inspector
D. Overland, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2011009-01	FIN (TBD)	Failure To Use Effective Engineering Controls As Part Of Pre-Job Planning To Reduce Contamination And Subsequent Exposure

Closed

05000382/20100005-02	URI	Removal of Radioactivity from the Steam Generators (Section 4OA5)
05000382/20100005-03	URI	Leakage from the Reactor Coolant Pump Seals (Section 2RS02)

LIST OF DOCUMENTS REVIEWED

Section 2RS02: Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
EN-RP-110	ALARA Program	6
EN-RP-110-01	ALARA Initiative Referrals	0
EN-RP-110-02	Elemental Cobalt Sampling	0
EN-RP-141	Job Coverage	5
N-RP-143	Source Control	7
EN-FAP-RP-001	Corporate ALARA Committee	1
HP-001-114	Control of Temporary Shielding	11

CONDITION REPORTS

CR-WF3-2009-05886 CR-WF3-2009-06033 CR-WF3-2009-07262 CR-WF3-2010-01745

RADIATION WORK PERMITS

20090508	Inspect/Rework RCP Motors 1B, 2A, 2B to include support work in shrouds
20090511	Work in Bowls (Eddy Current and Stage Equipment)
20090513	RCP 1A Motor and Driver Mount Removal and Replacement
20090600	HP Surveys/Roving Job Coverage in the Reactor Containment Building
20090601	Containment Coordination for Rigging and Support
20090606	Perform Minor Maintenance activities, walkdowns, surveillances, and inspections, NO work on valves, RCP motors/pumps, or the Reactor Head
20090610	Erect/Dismantle Scaffolding in the Reactor Containment Building
20090617	Radwaste Decon and Housekeeping
20090618	Remove/Replace Insulation in the Reactor Containment Building
20090702	Disassembly of Reactor Head and Associated Work Activities
20090705	Reassembly of Reactor Head and Associated Work Activities including Staging/Destaging of Equipment
20090707	Remove/Replace the ICI guide tubes (Thimbles)
20090708	ICI Removal/Installation to include cut up of ICIs and work on ICI Equipment

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE
QA-14/15-2009-W3-1	Quality Assurance Audit Report
	Quality Oversight Observations of Radiation Protection

MISCELLANEOUS DOCUMENT

TITLE	DATE
2009-2013 WF3 Five Year ALARA Plan	April 27, 2009
Preliminary Communication of Possible White Violation for Radiation Protection Inspection.	November 30, 2010
Supplemental Communication of Possible White Violation for Radiation Protection Inspection	December 17, 2010
Response to Supplemental Questions	February 22, 2011
Waterford 3 Response to NRC Questions	March 14, 2011
Waterford 3 RCP Leakage Flow Experience	March 17, 2011
Untitled (Clarification on the information provided December 17, 2010)	April 14, 2011

WHITE PAPER
PRELIMINARY COMMUNICATION OF POSSIBLE WHITE VIOLATION FOR RADIATION PROTECTION INSPECTION

1.0 Introduction

The NRC conducted a Radiation Protection (RP) Inspection the week of November 15, 2010, at Waterford 3 (WF3). On Thursday of inspection week the Lead Inspector informed the site of a possible White Violation for Occupational Radiation Safety based on NRC Significance Determination Process (SDP) IMC 0609, Appendix C. The site response to specified plant equipment or design issues was identified as the potential performance deficiency that affected collective dose. We understand the communication is preliminary and that further review and evaluation by the NRC is in progress regarding the basis and significance of the performance deficiency.

This position paper is intended to provide additional information for consideration by the NRC.

2.0 Issue

During the inspection, the NRC reviewed Radiation Work Permit (RWP) work packages where the collective dose was 5 Rem or greater and where the actual dose received was 50% greater than the original ALARA planned dose. Based on review of the RWP work packages, the NRC Lead Inspector communicated at the end of week brief (no exit meeting was conducted) that there were no ALARA planning or work control performance deficiencies associated with the RWP work packages.

The NRC Lead Inspector identified specific plant equipment or design issues that resulted in plant conditions which affected the collective dose of multiple work activities during RF-16. These five issues were:

- Fuel Failures
- Secondary Moisture Separator Reheater (MSR) Safety Relief Valve Opening with Subsequent Manual Reactor Trip
- Damaged Incore Instrument (ICI) Thimble
- Reactor Coolant Pump (RCP) Vapor Seal De-Staging and Leakage onto Insulation
- RCP Seal Heat Exchanger Gasket Leakage

We understand that the question is whether these plant conditions were reasonably anticipated and whether WF3 made prudent decisions and took reasonable effort to achieve ALARA given that the WF3 Three Year Collective Radiation Dose Average (TYRA) (years 2007, 2008, 2009) is greater than 135 person-rem and that there were more than 4 work activities which were affected by the plant equipment or design issues.

3.0 Conclusion

The dose impact due to the plant equipment and design issues were not directly related to the planning and controlling of individual radiological work activities and should not be attributed to ALARA performance deficiencies as causal factors.

The inclusion of the dose associated with unexpected changes as intended (planned) dose is described by example in NRC IMC 612, Inspection Reports, Appendix E, "Examples of Minor Issues", example "j". Consistent with this example, reasonably unexpected changes in radiological conditions from the time of initial ALARA job planning should be considered as intended dose instead of unintended dose. A change in scope beyond the control of associated job activities may warrant re-planning and the revised intended dose should be used determine if the 150% criteria was exceeded. Based on a review of the RWP packages, it has been determined that, for the population of RWPs in this review, the actual dose will be < 150% over the ALARA dose goal considering, as allowed in inspection guidance, the radiation dose resulting from reasonably unexpected conditions that were beyond the control of associated job activities.

The specific plant equipment or design issues that resulted in plant conditions which affected the collective dose of multiple work activities during RF-16 are listed with associated causes.

- Fuel Failures – this is a latent design issue associated with grid to rod fretting.
- Secondary Moisture Separator Reheater (MSR) Safety Relief Valve Opening with Subsequent Manual Reactor Trip – this was due to a manufacturing defect that caused an instantaneous failure.
- Damaged Incore Instrument (ICI) Thimble – this was a design issue which caused thimble elongation due to neutron fluence.
- Reactor Coolant Pump (RCP) Seal De-Staging and Leakage onto Insulation – this was a design issue associated with vapor seal.
- RCP Seal Heat Exchanger Gasket Leakage – this was a design configuration control issue.

The additional information supplied in Sections 5.0 and 6.0 documents that WF3 has been aggressive in establishing and maintaining a high industry standard in ALARA practices, both in the planning and work controls associated with radiological work as well as the management and decisions regarding plant equipment and design issues.

The prudent decision by Waterford 3 in 2007 to conduct a mid-cycle outage for inspections of the Steam Generators raised the current WF3 Three Year Collective Radiation Dose Average (TYRA) from 131.534 person-rem to 136.478 person-rem.

4.0 Regulatory Perspective

4.1 Determination of Work Activities

All management decisions regarding plant equipment and design issues should, and do, apply ALARA management principles. These issues are not related to the planning and controlling of individual radiological work activities and should not be attributed to ALARA performance deficiencies as causal factors in the scope of IP 71124.02, Occupational ALARA Planning and Controls.

The two NRC documents that provide specific requirements for the scope and basis for ALARA performance deficiencies are NRC Inspection Procedure 71124.02, Attachment 02, "Occupational ALARA Planning and Controls" and Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspection scope and performance deficiency criteria are addressed as specific work activities.

IMC 0609, Appendix C, defines work activity as, "One or more closely related tasks that the licensee has (or reasonably should have) grouped together as a unit of work for the purpose of ALARA planning and controls. In determining a reasonable grouping of radiological work, factors such as historical precedence, industry norms, and special circumstances should be considered."

As stated in NRC Inspection Procedure 71124.02, Occupational ALARA Planning and Controls, (03.02 Radiological Work Planning, a.), "A work activity is one or more closely related tasks that the licensee has reasonably grouped together as a unit of work for the purpose of ALARA planning and work controls. The effectiveness of a licensee's ALARA program is assessed by comparing the outcomes (in terms of collective dose) to the dose that was intended (i.e., determined to be ALARA) for individual work activities." The inspection parameters in 71124.02 include radiological work planning, dose estimates, source term reduction and control, and radiation worker performance.

Neither of these NRC documents includes site response to plant equipment or design issues as being within the scope of the ALARA inspection or as providing a basis for an ALARA performance deficiency.

Therefore, for the purposes of assessing ALARA performance, the scope of this assessment should not include as causal factors design and equipment issues that resulted from actions and decisions that were not related to the ALARA planning and controls of radiological work activity, particularly where there were no reasonable alternatives at the time of such prior actions and decisions that were far beyond the planning of individual work activities.

4.2 Inclusion of Revised Dose Estimates

In the potential white finding, radiological dose associated with outage jobs (13 RWPs) was categorized as more than a minor issue based on actual doses received exceeding 50% of the original ALARA planned (intended) dose. This did not take into consideration the revised (re-planned) dose estimates that included planned (intended) radiation doses resulting from reasonably unexpected conditions and were beyond the control of associated job activities as allowed in inspection guidance.

NRC IMC 0609 Appendix C, "Occupational Radiation Safety Significance Determination Process" and IMC 308, Att. 3, Appendix C, the associated technical basis, both state that "Collective dose associated with reasonably unexpected changes in the scope of work, material conditions, or radiological conditions, during a work activity (and for which measures are implemented to track, and if necessary, to reduce these doses) should also be considered intended dose."

The inclusion of the dose associated with unexpected changes as intended (planned) dose is described by example in NRC IMC 612, Inspection Reports, Appendix E, "Examples of Minor Issues", which provides examples for clarifying what types of issues are considered of minor significance. In reference to example "j", it is clear that a change in scope beyond the control of associated job activities may warrant re-planning and with a revised intended dose by which to determine if the 150% criteria was exceeded. Reasonably unexpected changes in radiological conditions from the time of initial ALARA job planning should be considered as intended dose instead of unintended dose.

The effects of recalculating the factor of unintended dose associated with the jobs that resulted in actual doses of 5 Rem or greater are illustrated in the summary table at the end of Section 6.0. The unintended dose for the 13 RWP work packages becomes planned (intended) dose The actual dose does not exceed 150% of the intended dose (potential white finding criteria) for any of the RWP work packages under consideration.

When utilizing the regulatory inspection guidance, none of the 13 RWP work packages meet the criteria for a performance deficiency.

Additionally, we believe that if a performance deficiency is determined to exist, its significance would be a minor violation based on the above regulatory guidance for inclusion of collective dose due to unexpected changes (within specific rules) as intended, or planned, dose.

4.3 2007 Mid-Cycle Steam Generator Inspection

In discussion with the NRC, Waterford 3 made a conservative decision to conduct a mid-cycle outage in 2007 to perform inspections of the Steam Generators. A total of 14.838 rem (DLR) was attributed to the 2007 mid-cycle outage. If this dose was not incurred, the total 2007 on-line dose of 5.287 rem (DLR) would be used as the 2007 reported dose making the TYRA 131.534 rem.

5.0 Technical Discussion of Issues

5.1 Fuel Failures

WF3 Transition from Standard Fuel to Next Generation Fuel

Discussion:

Actions taken to eliminate fuel failures by Cycle 17 were successful and reduced the overall dose impact by limiting the mixed core affect to one fuel cycle rather than two. Additionally, significant ALARA benefit is achieved by no fuel failures in Cycle 17, supporting dose reduction in RF-17.

Description of the Problem:

During Cycles 7 – 16, Waterford experienced failures in third cycle core periphery assemblies. Additionally, during Cycles 13, 14, and 15 second cycle fuel assemblies were located on the core periphery and failures were experienced in twelve (12) of these assemblies. Cycle 16 resulted in nine (9) second cycle fuel assembly leakers and two (2) third cycle fuel assembly leakers. Current RCS Chemistry data does not indicate any fuel failures for Cycle 17.

Actions were initiated several years ago (see timeline) to correct the known fuel failure mechanism (grid-to-rod fretting on core peripheral fuel assemblies). As part of these actions, inspections of Next Generation Fuel (NGF) Lead Test Assemblies (LTA) were performed to ensure the desired results were obtained. The results of these inspections allowed the site to accelerate the planned implementation of a full core of NGF to preclude fuel failures. Based on the plant's outage schedule the decision was made to implement NGF in two fuel cycles rather than the originally planned three cycles. This decision was based on:

1. Limit mixed core affects to one cycle rather than two – Mixed Core failure issues were known phenomena. The decision was made to transition in two cycles instead of three cycles (with only one mixed core cycle rather than two) to limit dose due to fuel failures to only one cycle of mixed core.
2. Eliminate adverse impact of having 3rd cycle peripheral fuel – if the transition had been made in three cycles, Cycle 17 would have had standard fuel on the periphery with fuel failures caused by the mixed core (i.e., cross flow between assemblies). A significant number of failures in 3rd cycle peripheral fuel assemblies was expected to occur if this had been done.
3. Ensure that WF3 was failure free prior to Steam Generator Replacement (RF17).
    a. This was important to minimize RCS activity during Cycle 17 for on-line ALARA;
    b. Entering RF17 with no fuel failures provides significant ALARA benefits during the long outage with opening of the RCS for Steam Generator Replacement.

Timeline:

Date	Discussion
Cycle 14 (June 2005 – November 2006)	Next Generation Fuel (NGF) Lead Test Assemblies (LTA) installed in core for 1st cycle

Date	Discussion
RF-14 (November 2006)	NGF LTAs inspected with good results after one cycle of operation
Cycle 15 (December 2006 – April 2008)	NGF LTAs installed for 2nd cycle Based on good results of LTA inspection, decision made to implement full core of NGF by C17 (100 assemblies in C16, 117 in C17)
RF-15 (April 2008)	NGF LTAs inspected with good results after two cycles of operation
Cycle 16 (June 2008 – October 2009)	NGF LTAs installed for 3rd cycle
RF-16 (October 2009)	Full core NGF installed (100 assemblies from C16, 117 fresh assemblies) Final LTA inspections begin with good results after three cycles of operation
Cycle 17 (December 2009 – present)	Final LTA inspections complete – preliminary findings indicate good results after three cycles of operation Full core NGF operating failure free at 350 EFPD

Comparison of Original Timeline versus Accelerated Timeline:

Cycle	Original Timeline	Revised Timeline
C14	LTA	LTA
C15	LTA*	LTA
C16	LTA	Mixed
C17	Mixed	NGF
C18	Mixed	NGF
C19	NGF	NGF

* C15 – Based on positive results of LTA inspections, decision made to accelerate transition to NGF to facilitate:

- Failure free core by Steam Generator Replacement in RF17 to reduce dose.
- Full core of failure free fuel by 2010 (INPO initiative).

Additional Actions to Reduce RCS Activity Prior to RF16:

During Cycle 16 it was recognized that the high RCS activity due to fuel failures would cause additional dose during the refueling outage (RF-16). As a result, actions were initiated to reduce the RCS activity and minimize outage dose

- Power reduction to 80% and return to 100% 3 days prior to scheduled outage to maximize RCS cleanup and minimize post shutdown Iodine spike. This power maneuver was done based on previous experience and input from other sites that the failed fuel would release its activity following the return to 100%, allowing it to be removed from the RCS prior to shutdown.

- Additional time scheduled following shutdown for RCS cleanup – 30 hours of RCS cleanup was added to the RF-16 schedule.

Conclusion:

The actions described above eliminated fuel failures in Cycle 17 and reduced the overall dose impact by limiting the mixed core affect to one fuel cycle rather than two. Additionally, significant ALARA benefit is achieved by no fuel failures in Cycle 17, supporting dose reduction in RF-17.

5.2 Secondary MSR Safety Relief Valve opening with Subsequent Manual Reactor Trip

Moisture Separator Reheater Relief Valve Pilot Spring Cracking Documented by CR-WF3-2010-05469 to the Increased Steaming Documented by CR-WF3-2009-04916

Discussion:

The Moisture Separator Reheater (MSR) Relief Valve failed due to a manufacturing defect. The compressed spring dislocated as it failed (severed), this dislocation is instantaneous and does not have a precursor to provide response time. The inability to makeup lost steaming inventory to the secondary plant resulted in a manual reactor trip from full power just days prior to the scheduled start of RF-16. The resulting unplanned crud burst raised reactor coolant activity well above that planned for the start of the refueling outage.

Description of the Problem / Timeline:

The failure mechanism identified by CR-WF3-2009-05469 was pilot spring breakage due to cracks formed at spring manufacture. The metallurgist who performed the failure analysis indicated that, "A spring with a crack or cracks from the manufacturer is such a rare phenomenon that it is not considered reasonably possible. One would need advance knowledge that a crack exists to reasonably consider crack propagation."

The root cause evaluation identified an unrelated organizational and programmatic (O & P) issue based on the availability of industry operating experience. Evaluation of available operating experience identified different failure mechanisms (pilot spring stress corrosion cracking, steam cutting of pilot valve seat) from that which occurred at Waterford 3 (cracks from the manufacturer). The O & P evaluation stated "CR-WF3-2009-4916 was generated on 9/16/09 indicating that the RS-203B seat leakage had worsened. This CR was designated as a category C and assigned to Planning & Scheduling Outage Management. There was no ACE assignment or OE research performed. The only resulting action from this CR was adding refurbishment of RS-203B (WO 102051) to the outage scope (SCR #317). MSR relief valve leakage was accepted by the plant as a low level issue." The identified O & P issue was that the available operating experience was not used when establishing risk, which was considered a weakness. The context was that using available operating experience may have prompted additional actions. On evaluation, no additional actions were determined that would have predicted or changed the outcome. Adding the RS-203B refurbishment to RF-16 scope a month prior to its start (when the leakage worsened) ensured that the condition would not exist in Cycle 17.

Conclusion:

The failure mechanism identified by CR-WF3-2009-05469 was pilot spring breakage due to cracks formed at manufacture, which is a rare phenomenon that would not be considered reasonably possible. The organizational and programmatic issue is a different issue which may have prompted additional evaluation, but would not have changed the outcome..

5.3 Damaged ICI Thimble

WF3 ICI Thimble Issue (E13)

Discussion:

In 2002, In-Core Instrumentation (ICI) Thimble Growth was identified to exist beyond original design assumptions. Based on this, WF3 took actions to replace the thimbles in RF-16. During RF-16, determined E13 ICI was bent between the Thimble Support Plate and Fuel Alignment Plate – a first time occurrence in the industry. This was removed using tooling already staged for replacing all ICI thimbles.

Description of the Problem:

Thimble Replacement - The decision was made to move replacement of thimbles from RF15 to RF16 based on measurement data collected in RF13. This was evaluated to ensure the thimbles did not contact the bottom of the fuel assembly guide tubes during operation. RF16 was deemed a more appropriate time to replace the thimbles as RF16 had scheduled Reactor Coolant Pump work, and thimble replacement would be less of an impact on outage critical path.

Crushed thimble at core location E13 - No industry Operating Experience existed for a thimble being crushed between the Upper Guide Structure and the Fuel Alignment Plate. This was an industry first occurrence and could not be anticipated. After the condition was discovered, WF3 was provided with information by the vendor that another plant had experienced growth that extended some thimbles below the Fuel Alignment Plate, but this growth did not result in collapse of the thimble.

Timeline:

Date	Discussion
2002	SONGS experienced neutron fluence induced growth of ICI thimbles in excess of original design assumptions. WF3 identified susceptibility due to similar design.
2003	Measurements were performed on the ICI plate and upper guide structure IAW work order 20882 during RF12 and found to be within tolerances of its nominal position.
2003	Spacers installed in the ICI flanges during RF12 raised the thimble plate 3 inches to provide sufficient allowance for growth.

Date	Discussion
2005	Interim modification made to thimble support plate to allow for two cycles of operation before the thimbles needed to be replaced with shorter thimbles; actual measurements during RF13 showed the modification would allow three cycles of growth (i.e. until RF16).
2005	Decision made to replace thimbles in RF16 based on impact to outage durations and impact on other site work.
2007	Funding approved to replace ICI thimbles in RF16; decision made to have all necessary thimbles built and ready for RF15 as contingency.
2008	Startup from RF15 with E13 ICI not responding.
2009	During Reactor disassembly, discovered E13 ICI was bent between the Thimble Support Plate and Fuel Alignment Plate.
2009	Replaced all recoverable ICI thimbles.

Conclusion:

The WF3 plan to replace ICI thimbles was based on industry operating experience. Actions taken along the way ensured operability of the ICI system. ICI E13 being bent between the Thimble Support Plate and Fuel Alignment Plate during reactor re-assembly in RF15 was a first time occurrence in the industry and could not have been foreseen.

5.4 RCP Vapor Seal De-Staging and Leakage onto Insulation

RCP Vapor Stage "de-staging" Due to Quad Ring Hang Up

Discussion:

The timeline associate with the postponement of installing the vapor stage modification (EC 3093) until RF-17 was appropriate and conservative.

The N-9000 vapor stage modification was initially installed at WF3 during RF-12 (Fall 2003). The first vapor stage seal de-staging issue was observed during the September 2005 shutdown for Hurricane Katrina. Actions to correct this condition were initiated with the vendor.

The design change from the vendor (Flowserve) was not complete in time for installation in RF-15, and therefore not available for RF-15 shutdown. The modification is planned for upcoming outages. The risk of a installing a new design at multiple sites-site identified as both W3 and ANO would be implementing the upgraded vapor stage. Operating experience with the new design was limited. The decision to delay installation of the vapor stage seal modification supported obtaining ANO experience to validate the new seal performed as designed and did not create additional issues. Additionally, to reduce dose in future outages, WF3 installed a modification in RF-16 which captures the vapor stage leakoff and routes it to floor drains, mitigating the impact until the vapor stage seals are modified / upgraded.

Description of the Problem:

At Low Reactor Coolant System operating pressures, the N9000 vapor stage seal faces open to the Reactor Coolant Pumps. This separating of the seal faces causes excessive seal leakage in the RCP shroud area. This phenomenon was first observed during the Hurricane Katrina Shutdown, September 2005 (CR-WF3-2005-03831) and again during RF-14 (CR-WF3-2006-03597).

The opening of the seal faces is the result of the stationary face quad ring "hanging up" on the balance sleeve. If the stationary face quad ring "hangs up" or sticks to the balance sleeve, it prevents the stationary face from maintaining proper contact with the rotating face during relative movement of either component. The rotating face is free to move up and down with the pump shaft in response to temperature changes, such as heat-up or cool down, or up or down thrust of the shaft when starting or stopping the pump. The stationary face moves in response to changes in the rotating face position or due to changes in the pressure in the vapor stage pressure, or controlled bleed-off pressure.

The reason why this occurs is a vendor design change of the vapor stage seal. Flowserve, the current vendor of Byron Jackson mechanical seals developed a new seal design (N9000) and no longer carries the original seal design (SU Cup). The tendency for the quad ring to stick is an inadequate design of the N9000 seal design.

Currently scoped in RF-17 is to install the vapor stage modification EC 3093 on RCP 2A. Following RF-17 the vapor stage modification will be installed during pump rebuilds and future seal replacements on the remaining Reactor Coolant Pumps.

Timeline:

Date	Discussion
Fall 2001	ER-2001-0292 Initiated to Install N9000 Seal.
RF Fall 2003	N9000 Seal ER installed in RCP-1B.
RF Spring 2005	ANO 2 Experiences Vapor Stage leakage.
RF Spring 2005	N9000 Seal ER installed in RCP-2B.
Hurricane Katrina (September 2005)	CR-WF3-2005-3831 Vapor Stages open during plant shutdown.
September 2005	Flowserve contacted and describes mechanism vapor stage quad ring to hang up.
March 2006	Flowserve provides proposal to address vapor stage leakage.
RF-14 (November 2006)	CR-WF3-2006-03597 - RCP-2B Vapor Stage Opens during plant shutdown (vapor stage quad ring hang up).
January 2007	Funding (SIPD 354) approved for Flowserve to perform a study on possible solutions to vapor stage quad ring to hang up.
October 2007	CR-WF3-2007-03716 - Reactor Coolant Pump vapor stage leakage caused by vapor stage quad ring to hang up and vapor stage leakoff line not draining to Reactor Drain Tank.
RF-15 (April 2008)	Vapor Stage leakoff lines rerouted to a floor drain rather than the Reactor Drain Tank (EC 6256).
September 2008	Study Completed by Flowserve and recommended a modification to the vapor stage to increase the amount of spring force on the vapor stage faces to overcome quad ring to hang up

Date	Discussion
Prior to RF-16 (October 2009)	Critical Decision made to postpone installing the vapor stage modification (EC 3093) until RF-17 due to the risk of a multi-site threat to generation was identified because both W3 and ANO would be implementing the upgraded vapor stage. Additionally, operating experience with the new design was limited. Decision supported obtaining ANO experience to validate the new seal performed as designed and did not create additional issues. Installed trough modification (EC 18520) in RF-16 to mitigate the impacts until vapor stage seals modified / upgraded.
RF-16 (October 2009)	CR-WF3-2009-05501 - Boric Acid discovered in RCP 2B due to heat exchanger gasket leakage, vapor stage leakoff line not performing its design function, and quad ring hang up.
RF-16 October 2009)	EC 18520 installed to reroute the vapor stage leakoff line from each pump to individual floor drains, by passing the in line check valve.

Conclusion:

The timeline associated with the postponement of installing the vapor stage modification (EC 3093) until RF-17 was appropriate and conservative. The design change from the vendor (Flowserve) was not complete in time for installation in RF-15, and therefore not available for RF-15 shutdown.

For RF-16, the risk of a installing a new design at multiple sites-site identified that both W3 and ANO would be implementing the upgraded vapor stage. Operating experience with the new design was limited. The decision to delay installation of the vapor stage seal modification supported obtaining ANO experience to validate the new seal performed as designed and did not create additional issues. Additionally, to reduce dose in future outages, WF3 installed a modification in RF-16 which captures the vapor stage leakoff and routes it to floor drains, mitigating the impact until the vapor stage seals are modified / upgraded.

5.5 RCP Seal Heat Exchanger Gasket Leakage

Discussion

During the 2007 mid-cycle outage for steam generator inspections, Reactor Coolant Pump (RCP) 1A Heat Exchanger gasket leakage was found. The Heat Exchanger gasket was replaced with a thinner gasket and the original preload. No additional leakage was found during the subsequent refueling outage (RF-15, April 2008).

During RF-16 (October 2009) boric acid was discovered on RCP 2B. The root cause was determined to be inadequate design of the vapor stage seal leak-off line and possible heat exchanger gasket leakage due to low margin in the design of the heat exchanger bolted connection.

In June, 2010 a borescope inspection was performed on RCP 1B and 2A, with no visible boric acid buildup.

Description of the Problem

CR-WF3-2009-05501

The root cause of inadequate design of the vapor stage seal leak-off line was caused by the check valves installed on the leak-off lines were incapable of passing flow as intended by design. EC 18520 was implemented on all four Reactor Coolant Pumps to address this issue.

The possible root cause of heat exchanger gasket leakage due to low margin design of the heat exchanger bolted connection was determined to be caused by inadequate heat exchanger gasket compression at the joint which could result in leakage at the flange connection. This was validated by the Flowserve prepared Structural Analysis Report Reconciliation of Spec. No.: 9270-PE-480 (addendum 2009-1 SR1010-2). Both the original and SMP-1427 joint configuration designs were evaluated to have low margin. EC 18557 further improved gasket compression and was installed on RCP 1A and 2B. This design will be implemented on RCP 2A during RF-17 and RCP 1B during RF-18.

Date	Discussion
RF-01 (January 1987)	RCP Heat Exchanger gasket leaking. SMP-1427 was installed to prevent RCP Heat Exchanger gasket leakage (Thicker gasket installed with higher preload). SMP-1427 Affected Documents were not revised as Specified in SMP (RCP Tech Manuals and drawings were not revised to include the thicker heat exchanger gasket and higher preload).
RF-04 (April 1991)	RCP 2A overhauled due to outer casing gasket leakage. Heat Exchanger gasket replaced with thinner gasket and original preload.
July 1996	Heat Exchanger Scoring due to Baffle Bolts allowing Baffle to contact Heat Exchanger on RCP 2B. Heat Exchanger replaced with thinner gasket and original preload.
RF-10 (November 2000)	RCP 2B pump: replaced, thinner heat exchanger gasket and original preload used.
Cycle 15 Mid-Cycle Outage (October 2007)	RCP 1A Heat Exchanger gasket leakage is discovered. Heat Exchanger gasket replaced with thinner gasket and original preload.
RF-15 (April 2008)	No boric acid buildup observed on RCPs.
RF-16 (October 2009)	CR-WF3-2009-5501 Boric Acid discovered on RCP 2B Inadequate heat exchanger gasket and preload discovered. EC 18557 was implemented to increase the gasket thickness and heat exchanger stud preload. RCP 1A pump overhaul which included the replacement of the heat exchanger connection. EC 18557 was implemented to increase the gasket thickness and heat exchanger stud preload. Borescope inspection tubes installed for the RCPs during this outage.
June 2010	Containment entry performed to inspect RCP 1B and 2A with the borescope. No visible buildup of boric acid observed.

Conclusion

The root cause of inadequate design of the vapor stage seal leak-off line was resolved during RF-16 by implementation of EC 18520 on all four Reactor Coolant Pumps.

The possible root cause of heat exchanger gasket leakage due to low margin design of the heat exchanger bolted connection was resolved by implementing EC 18557 on RCP 1A and 2B during RF 16, and is currently scheduled for RCP 2A during RF-17 and RCP 1B during RF-18.

6.0 ALARA Decisions and Actions for Work Activities

6.1 Summary Table

This summary table illustrates the impact on each RWP when the radiation dose resulting from reasonably unexpected conditions that were beyond the control of associated job activities is adjusted as intended dose in accordance with NRC IMC 0609, Appendix C.

The subsequent work activity discussions also list the RWP data applicable to each.

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090508	Bulk RCP motor work	2.701	5.649	209	2.272	3.377	84	5.110	111
20090511	S/G eddy current	13.895	19.420	140	11.071	8.349	80	24.539	79
20090513	RCP 1A motor/pump replacement	**13.590**	**29.692**	**218**	19.601	10.091	144	28.396	105
20090600	HP in RCB	6.923	15.604	225	8.836	6.768	128	15.527	100
20090601	RCB Coordinators	2.664	6.271	235	3.478	2.793	131	5.443	115
20090606	Minor maintenance	4.944	10.446	211	6.629	3.817	134	8.638	121
20090610	Scaffolding in RCB	9.371	22.447	240	12.443	10.004	133	19.609	114
20090618	Insulation	3.166	8.667	274	3.144	5.523	99	7.544	115
20090702	Reactor disassembly	4.326	8.751	202	5.754	2.997	133	7.320	120
20090705	Reactor reassembly	5.626	16.320	290	8.120	8.200	144	13.677	119
20090707	ICI thimble modification	**13.053**	**28.339**	**217**	18.815	9.524	144	23.831	119
20090708	ICI withdraw and cut-up	2.648	8.579	324	3.735	4.844	141	8.173	105
20090503	RCP 1A, 1B, 2A & 2B Gutter Mod. Installation	4.120	6.027	146	5.427	0.6	132	6.12	98

Attachment 2

6.2 RWPs of Interest During NRC Inspection

2009-0508, Inspect/Rework RCP Motors 1B, 2A and 2B to include support work in shrouds (spool piece work, oil pan work, pie plates, etc.)

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090508	Bulk RCP motor work	2.701	5.649	209	2.272	3.377	84	5.110	111

Dose estimates for this job were revised as follows:

- From 2.701 rem to 3.901 rem (revision 3)
- From 3.901 rem to 5.110 rem (revision 4)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Expanded scope due to discovery during outage (motor conduit)	3.000
Higher than planned dose rates	.377
Total	**3.377**

Changes Made/Dose mitigation During Performance of Work

- Removed and replaced insulation saturated from leaking RCP seals
- Decontaminated around each RCP

2009-0511, Eddy Current Work/Tube Plugging Inside of the Steam Generators Primary Side and Equipment Staging/de-staging

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090511	S/G eddy current	13.895	19.420	140	11.071	8.349	80	24.539	79

Dose estimates for this job were revised as follows:

- From 13.895 rem to 24.539 rem (revision 3)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Leaking RCP vapor stage seals; hard reactor trip from 100% power and CRUD plate-out	.242
CRUD Plate-out in RCS Piping from hard reactor trip (D-rings)	8.107
Total	**8.349**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

2009-0513, Reactor Coolant Pump 1A Motor and Driver Mount Removal and Replacement

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (rem) (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090513	RCP 1A motor/pump replacement	**13.590**	**29.692**	**218**	19.601	10.091	144	28.396	105

Dose estimates for this job were revised as follows:

- From 13.590 rem to 19.454 rem (revision 1)
- From 19.454 rem to 28.396 rem (revision 3)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	**Dose (rem)**
Discovery of light erosion pattern on rotating baffle	.017
Discovered significant wear on RCP 1A motor lower oil cooler return piping	.202
Seismic link field fit-up issues due to unforeseen clearance conditions	.822
RCP 1A main power cable fit-up due to unforeseen manufacturing issue	.242
Discovery of vendor phase rotation issue (motor)	.278
High RCS activity from hard reactor trip, high dose rates on insulation packages from RCP vapor seal de-staging	6.030
Driver mount/volute housing separation due to unforeseen thermal binding	2.500
Total	**10.091**

Changes Made/Dose mitigation During Performance of Work

- Removed and replaced insulation saturated from leaking RCP seals
- Decontaminated around each RCP
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

2009-0600, HP Surveys/Roving Job Coverage in the Reactor Containment Building and Installation/Removal of RADS in Containment

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090600	HP in RCB	6.923	15.604	225	8.836	6.768	128	15.527	100

Dose estimates for this job were revised as follows:

- From 6.923 rem to 7.913 rem (revision 2)
- From 7.913 rem to 12.413 rem (revision 3)
- From 12.413 rem to 15.527 rem (revision 4)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Leaking RCP vapor stage seals/CRUD Plate-out in RCS Piping from hard reactor trip (D-rings)	4.528
Extension of outage from planned 35 days to 46 days	2.240
Total	**6.768**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

2009-0601, Reactor Containment Building/Fueh Handling Building and including Lay-down Areas Outside Containment

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090601	RCB Coordinators	2.664	6.271	235	3.478	2.793	131	5.443	115

Dose estimates for this job were revised as follows:

- From 2.664 rem to 5.443 rem (revision 4)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Leaking RCP vapor stage seals/CRUD Plate-out in RCS Piping from hard reactor trip (D-rings)	2.793
Total	**2.793**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from vapor seal stage on RCPs

2009-0606, Entries into Posted Radiation and High Radiation Areas of the Reactor Containment Building to Perform Minor maintenance Activities, Walkdowns, Surveillances and Inspections.

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090606	Minor maintenance	4.944	10.446	211	6.629	3.817	134	8.638	121

Dose estimates for this job were revised as follows:

- From 4.944 rem to 5.888 rem (revision 1)
- From 5.888 rem to 8.638 rem (revision 2)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Leaking RCP vapor stage seals/CRUD Plate-out in RCS Piping from hard reactor trip (D-rings)	2.013
Extension of outage from planned 35 days to 46 days	1.804
Total	**3.817**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

2009-0610, Erect/Dismantle Scaffolding in the Reactor Containment Building

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090610	Scaffolding in RCB	9.371	22.447	240	12.443	10.004	133	19.609	114

Dose estimates for this job were revised as follows:

- From 9.371 rem to 13.083 rem (revision 2)
- From 13.083 rem to 19.609 rem (revision 3)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Leaking RCP vapor stage seals/CRUD Plate-out in RCS Piping (D-rings)	5.846
Expanded scope/discovery (unforeseen scaffold installations and adjustments)	4.158
Total	**10.004**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

<u>2009-0618, Remove/Replace Insulation in the Reactor Containment Building</u>

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (rem) (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090618	Insulation	3.166	8.667	274	3.144	5.523	99	7.544	115

Dose estimates for this job were revised as follows:

- From 3.166 rem to 7.544 rem (revision 2)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	**Dose (rem)**
Leaking RCP vapor stage seals/CRUD Plate-out in RCS Piping (D-rings)	1.214
Expanded scope/discovery	4.309
Total	**5.523**

Changes Made/Dose mitigation During Performance of Work

- Removed contaminated insulation blankets on RCPs 1A, 1B, 2A and 2B
- Decontamination around RCPs
- Installed additional temporary shielding on Safety Injection Lines and on -11 RCB to shield dose rates from leaking RCS from RCPs

2009-0702, Disassembly of Reactor Head and All Associated Work Activities

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090702	Reactor disassembly	4.326	8.751	202	5.754	2.997	133	7.320	120

Dose estimates for this job were revised as follows:

- From 4.326 rem to 5.126 rem (revision 1)
- From 5.126 rem to 5.407 rem (revision 2)
- From 5.407 rem to 6.074 rem (revision 3)
- From 6.074 rem to 6.395 rem (revision 5)
- From 6.395 rem to 7.320 rem (revision 6)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Elevated dose rates in cavity from failed fuel and E-13 thimble	1.147
Discovery of existing damage of CEA extension shaft and subsequent removal	.697
Discovery of stuck Reactor Vessel studs	1.153
Total	**2.997**

Changes Made/Dose mitigation During Performance of Work

- Decontaminated refueling canal

2009-0705, Reassembly of Reactor Head and all Associated Work Activities

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090705	Reactor reassembly	5.626	16.320	290	8.120	8.200	144	13.677	119

Dose estimates for this job were revised as follows:

- From 5.626 rem to 10.677 rem (revision 2)
- From 10.677 rem to 13.677 rem (revision 4)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Elevated dose rates in cavity from failed fuel and E-13 thimble	3.562
Added scope (not originally planned) reactor vessel closure head stud elongation check	2.333
Installation of studs for head vent line flange	.255
ICI flange assembly – (added scope for seal carriers)	2.050
Total	**8.200**

Changes Made/Dose mitigation During Performance of Work

- Decontaminated refueling canal

2009-0707, Damaged ICI Thimble (E-13)

Pre-outage planning

Pre-Refuel 16 planning did not consider a bent In-Core Instrument. This condition was self revealing once the ICI thimble project was under-way and could not have been foreseen in the ALARA Planning of the RWP. This was emergent work during Refuel 16.

Refuel 16 Performance

RWP 2009-0707 was generated pre-outage to support work for the ICI thimble modification. As listed in the table below, the initial dose estimate approved by the ALARA Manager's Committee was 13.053 rem and the actual dose received for the project was 28.339 rem.

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090707	ICI thimble modification	**13.053**	**28.339**	**217**	18.636	9.703	143	23.831	119

This RWP was re-planned to 23.831 person-rem.

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Difficulty in accessing hand rail locations to be removed due to inability to obtain data pre-outage	.312
FME and vacuum debris from grinding interferences on UGS (E-13 thimble)	.110
Implemented EDM contingency plan	2.636
Emergent E-13 thimble work	2.250
Elevated dose rates for E-13 (due to location)	.526
Unforeseen interference removal for cheese plate	.179
Elevated contamination levels on ICI thimble rack due to E-13 fragments and high RCS activity from failed fuel in the refuel canal	3.690
Total	**9.703**

Changes Made/Dose mitigation During Performance of Work

- RWP reviewed by ALARA Manager's Committee.
- Added an additional Tri-Nuc unit to the refuel canal to assist in removing particulate material from E-13 removal and grinding activities
- Installed additional temporary shielding on the auxiliary bridge
- Used plant expertise from Arkansas Nuclear One to provide input

2009-0708, ICI Removal/Installation to Include Cut-up of ICIs and Work on ICI Equipment

A	B	C	D	E	F	G	H	I	J
RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	Initial % Delta (D/C)	Justified Dose (rem) (D-G)	Justification - Based on RWP documentation	new % Delta (using justified dose) (F/C)	Final approved RWP Dose Estimate (rem)	Actual dose to Final RWP dose Estimate (%) (D/I)
20090708	ICI withdraw and cut-up	2.648	8.579	324	3.735	4.844	141	8.173	105

Dose estimates for this job were revised as follows:

- From 2.648 rem to 4.573 rem (revision 1)
- From 4.573 rem to 8.173 rem (revision 3)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Elevated dose rates in cavity (E-13)	4.190
Equipment failures (ICI cutter)	.526
Emergent work (stuck ICI)	.128
Total	**4.844**

Changes Made/Dose mitigation During Performance of Work

- Decontaminated refueling canal
- Installation of an additional Tri-Nuc unit
- Installation of shielding on the auxiliary refuel bridge

2009-0503, RCP 1A, 1B, 2A, 2B Gutter Installation and associated support activities

This RWP was not pre-planned prior to Refuel 16. This job was a result of dose mitigation to minimize RCP seal leakage when running RCPs during plant start-up. This modification was effective in maintaining the RCP seal leakage to the trough area and not re-saturating the newly-installed insulation blankets on each of the four RCPs.

This modification is a first time evolution.

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (rem) (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090503	RCP 1A, 1B, 2A & 2B Gutter Mod. Installation	4.120	6.027	146	5.427	0.600	132	6.12	98

Dose estimates for this job were revised as follows:

- From 4.120 rem to 6.120 rem (revision 1)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Unforeseen interference when insulation panel was adjusted (field fit-up)	0.600
Total	**0.600**

6.3 Summary: ALARA Decisions and Actions for Work Activities

The population of RWPs documented in this review will be < 150% over the initial (pre-planned) dose goal when the dose impacts of the five conditions below are considered and the radiation dose resulting from reasonably unexpected conditions that were beyond the control of associated job activities are adjusted as intended dose in accordance with NRC IMC 0609, Appendix C.

1. Fuel failures and effect on Refuel 16 work
2. Early CRUD burst and plate out due to manual reactor trip
3. Damaged ICI Thimble (E-13)
4. RCP Vapor Seal Leakage
5. RCP Seal Heat Exchanger Gasket Leakage.

WHITE PAPER
SUPPLEMENTAL COMMUNICATION OF POSSIBLE WHITE VIOLATION FOR RADIATION PROTECTION INSPECTION

Section 1.0 Introduction

The NRC conducted a Radiation Protection (RP) Inspection the week of November 15, 2010, at Waterford 3 (WF3). The NRC informed the site of a possible White Violation for Occupational Radiation Safety based on NRC Significance Determination Process (SDP) IMC 0609, Appendix C. A teleconference was held at 10:30 AM on 12/01/2010 to discuss the white paper that had been forwarded to the NRC titled: PRELIMINARY COMMUNICATION OF POSSIBLE WHITE VIOLATION FOR RADIATION PROTECTION INSPECTION.

Plans were discussed at the meeting to prepare a timeline of the operational condition of the plant from the pre-outage downpower on 10/17/09 to removing the Steam Generator hot and cold leg manways on 11/5/09. This period overlaps the conditions of interest. Control Room Logs, RCS level, RCS pressure, and RCS activity levels were included in the package that was provided to the NRC on 12/10/10.

Section 2.0 Performance Deficiency

During the 12/1/10 conference call, the NRC clarified the potential performance deficiency as: Waterford 3 failed to meet procedurally directed goals in CE-002-006, Section 10.8 "Reactor Coolant System Chemistry Control for Refueling" in cleaning up the Reactor Coolant System (RCS), which led to excessive dose in various jobs.

Subsequent discussion indicated that an additional factor being evaluated is whether the outage schedule unduly impacted Waterford 3's decisions associated with RCS cleanup.

Section 3.0 Conclusion

No ALARA performance deficiency exists. As demonstrated by the included discussion:

1) Waterford 3 had a pre-outage plan in place that appropriately addressed dose reduction to achieve ALARA objectives. The site's activities taken pre-outage demonstrate that management was aware of the radiological challenges for controlling dose and appropriate plans were developed for the issues that were foreseeable. The plan included actions to address potential issues resulting from elevated Iodine and noble gas due to fuel leaks, as experienced at other plants. (This is further discussed in section 4.0.)

2) Waterford 3 complied with the requirements of procedure CE-002-006, Maintaining Reactor Coolant Chemistry, Section 10.8 "Reactor Coolant System Chemistry Control for Refueling." Specifically, the RCS cleanup requirements of CE-002-006, Section 10.8, Step "m". (This is further discussed in section 5.0.)

3) Waterford 3 actions during the outage appropriately balanced nuclear and radiological safety. Outage decisions were consistent with ALARA principles and were not unduly impacted by the outage schedule. For example, 38 hours were added to the outage scope during the outage to provide additional cleanup time to lower dose. Outage planning precluded the higher risk associated with installation of SG nozzle dams at reduced RCS inventory while SDC is needed to remove decay heat from the fuel in the reactor vessel. (This is further discussed in section 6.0.)

4) Installation of the vapor seal modification could not be done prior to RF16 because the design and fabrication of the modification had not been completed. (This is further discussed in section 7.0.)

Section 4.0 Pre-Outage Planning

During Cycle 16 it was recognized that high RCS activity due to fuel failures would cause additional dose during the Refueling Outage (RF16). As a result, actions were initiated to reduce RCS activity and minimize outage dose. The establishment of dose reduction activities included the following:

- A power reduction to approximately 80% and return to 100% was performed 3 days prior to the scheduled outage to maximize RCS cleanup and minimize the post shutdown Iodine spike. This power maneuver was performed based on operating experience and input from other sites that the failed fuel would release activity following the return to 100%, allowing it to be removed from the RCS prior to shutdown.

- Prior to Refuel 16, RCS Iodine-131 and Xenon-133 gas clean up activity levels were established. The goals were based on 0.3 DAC for Iodine and .03 DAC for Xenon, in the containment, with anticipated full destaging of the Reactor Coolant Pump Seals vapor stages at approximately 400 psi. This translates into an RCS Iodine-131 activity of 0.0045 uCi/ml and RCS Xenon-133 activity of 0.16 uCi/cc. The cleanup goals were captured in the Waterford 3, Refuel 16, Reactor Coolant System Dose Equivalent Iodine and RCS Degassing Plan. There were 30 hours placed in the Refueling Outage for reactor coolant system Iodine-131 and Xenon-133 cleanup.

- Cobalt cleanup plans were developed which included a forced liberation of hard gamma isotopes at acid oxidizing conditions followed by running two reactor coolant pumps, one in each loop, until the Cobalt peaked. Once the peak was reached, plans included cleanup to the hard gamma target value of 0.05 uCi/ml prior to cavity flood up (which was met as reflected in Section 5.0 of this paper). The hard gamma activity target value of 0.05 uCi/ml was in place during previous outages and is in compliance with EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines. The hard gamma target value is also contained in procedure CE-002-006, Maintaining Reactor Coolant Chemistry.

  This plan was developed with the knowledge that in order to obtain a forced liberation of hard gamma isotopes by use of Hydrogen Peroxide, RCS temperature should be below 200 degrees F. The Shutdown Cooling System (SDC) is the only means available to achieve this required temperature. RCS pressure must be below 392 psia before the SDC system can be placed in service. Thus, the plant design creates plant operating conditions where the forced liberation occurs at RCS pressure where RCP vapor seal destaging may be also occurring. That is, leakage of high activity water from the RCP seals cannot be avoided. Additionally, at this low RCS pressure, letdown flow is reduced from approximately 120 GPM to approximately 80 GPM, which reduces the rate of cleanup. Securing the RCPs shortly after the hydrogen peroxide addition and Cobalt peak is required for placing the higher flow rated Shutdown Cooling System Purification System in service. The Shutdown Cooling Purification System can provide approximately 230 gpm purification flow.

- The pre-outage plan directed that the plant would reduce pressure to between 1000 and 1200 psia to maintain control of letdown flow and provide margin to the destaging pressure of the RCP Seals and minimize vapor seal leakage. Waterford 3's design creates plant operating conditions where peroxide addition occurs at an RCS pressure where RCP vapor seal destaging is also occurring. Therefore, leakage of high crud content water from the RCP seals can occur. Compounding this, as RCS pressure lowers, letdown flow lowers, which reduces the rate of cleanup. Until, at atmospheric pressure, there is insufficient pressure to support coolant flow through the letdown system purification system. It has been Waterford 3's experience that the seals do not substantially destage until RCS pressure goes < 400 psia upon securing of the

RCPs. This plan was successful and there was no apparent leakage from the seals at a pressure of 1100 psia for RF16.

- Operations is procedurally directed to align both letdown flow control valves and both backpressure control valves at 1200 psia; this supports additional letdown flow for RCS cleanup at a lower motive pressure from the RCS.

Section 5.0 Compliance with Procedures

Chemistry goals for RCS cleanup are located in CE-002-006, Maintaining Reactor Coolant Chemistry and contains the following criteria in Section 10.8:

Continue cleanup of RCS until the following recommended target values are met:

- *I-131 < 0.01 uCi/ml*
- *Xe-133 < 0.5 uCi/cc*
- *Hard Gamma Emitters (Co-58 + Co-60 + Cs-134 + Cs-137 + Mn-54) < 0.05 uCi/ml*

These Reactor Coolant System Chemistry target values were achieved as follows:

I-131	0.00887 uCi/ml (10/26/09 16:01 per Chemistry logs)
Xe-133	0.206 uCi/cc (10/23/09 18:28 per Chemistry logs)
Hard Gamma	0.0375 uCi/ml (10/26/09 16:01 per Chemistry logs)

The target for RCS Xe-133 concentration was achieved on 10/23/09 while CVC purification was being used prior the flood up. Because the RCS needs to be depressurized in order to continue to release gases from the fuel for continued cleanup, letdown flow to burp the VCT is not available to continue degassing Xe-133 from the RCS. Whereas Iodine can be removed by ion exchange, Xenon cannot. Since the fuel continues to release its Xe-133, it is unavoidable that Xe-133 concentration will rise. Upon filling the Refueling Cavity, Xe-133 was 0.0567 uCi/cc. It is notable that the Refueling Cavity Xe-133 activity was not a major contributor to dose because Xe-133 is a significantly lower energy gamma when compared to Co-58.

The Refueling Water Storage Pool was cleaned to the following specifications prior to use of the water for flooding the refueling cavity, based on data recorded for 10/25/2009 08:55.

I-131	0.000957 uCi/ml
Gross Activity	0.0386 uCi/ml
Hard Gamma	0.00560 uCi/ml

Containment Air Samples taken during the Refueling Cavity Fill confirmed that the containment atmosphere was below the 0.5 DAC limit.

10/27/2009 15:10	Containment Activity 0.15 DAC
10/27/2009 23:47	+21 Cont Atmosphere: 0.3 DAC; -40 Cont Atmosphere: 0.2 DAC
10/29/2009 03:42	+21 Cont Atmosphere: 0.47 DAC

Waterford 3's procedures use the following methodology for performing a forced liberation of hard gamma isotopes for refueling preparations. This methodology follows the guidance in the EPRI Pressurized Water Reactor Primary Water Chemistry Guideline. The ultimate goal is to have the reactor coolant system in an acidic oxidizing environment to bring plated Cobalt into solution for cleaning.

1. Addition of boric acid places the reactor coolant system in an acidic condition. This is a favorable condition for placing Cobalt into solution vs. a particulate form.

2. Nitrogen is placed on the volume control tank to lower hydrogen in the reactor coolant system. This allows the addition of hydrogen peroxide for chemical degassing and oxidation of Cobalt species.

3. Shutdown Cooling (SDC) is placed into service. This allows for cooling of the reactor coolant system to allow for peroxide addition. This requires securing two of four reactor coolant pumps for cooldown. A limited amount of letdown system purification flow (approximately 80 gpm) is available.

4. Hydrogen peroxide is added to oxidize Cobalt in the reactor coolant system.

5. Once a Cobalt peak is reached, the remaining two operating reactor coolant pumps are secured.

6. RCS pressure is lowered to atmospheric to: 1) release the remainder of fission product gases so that they can be cleaned from the system, and 2) establish SDC purification flow at up to 230 gpm. At atmospheric pressure, no letdown system purification flow is available.

7. Once Chemistry determines that the 0.05 uCi/ml hard gamma target value is reached, the reactor cavity may be flooded from the refueling water storage pool.

CE-002-006 Section 10.8 and the EPRI guideline both allow securing of the Reactor Coolant Pumps (RCPs) following the addition of hydrogen peroxide followed by RCS cleanup to target values via the shutdown cooling purification system.

The sequence followed during this outage up to placing SDC purification in service, including the timing of securing the RCPs following the Cobalt peak, has been the approach taken in the past. The sequence is the methodology described above. This methodology follows the guidance in the EPRI Pressurized Water Reactor Primary Water Chemistry Guideline. The flood up target values in procedure CE-002-006 of 0.05 uCi/ml hard gamma and 0.01 uCi/ml I-131 are taken directly from the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines.

The EPRI Pressurized Water Reactor Primary Water Chemistry Guideline, Revision 6, page 3-26, note 1, suggests that the hard gamma activity $\leq$ 0.05 uCi/ml and I-131 $\leq$ 0.01 uCi/ml target values of the RCS should be met prior to filling the cavity to assure that the dose one meter above the surface of the refueling water is less than 5 mR/hr. Both of these Reactor Coolant activity target values were met prior to filling the cavity.

The EPRI guideline does not provide a flood up target value for Xe-133. CE-002-006 contains an additional target value for Xe-133 < 0.5 uCi/cc, which was met as reflected above.

During Refuel, 16 the procedurally required targets for source term cleanup were met. In fact, extensive efforts were taken to reduce the source term as demonstrated by the 50 hours spent on hard gamma cleanup to 0.05 uCi/ml. the plant was prudently operated to in accordance with procedures to reasonably reduce dose.

Section 6.0 Outage Execution

Management decisions regarding plant equipment and design issues should and do apply sound ALARA management principles. Waterford 3 believes that reasonable decisions were made for RF16 relative to dose reduction. Actions taken for RCS cleanup include the following:

While shutdown and as the RCS was depressurized, more Iodine in the fuel pins came into equilibrium with the RCS. The plant needed to depressurize to 800-850 psia to continue motivating fission products from the leaking fuel into the RCS for advancing the cleanup. Planned cleanup continued for 20 hours at this pressure.

Following the unanticipated hard Reactor Trip, Waterford 3 proactively added an additional 38 hours to the outage schedule for RCS cleanup. Once at atmospheric pressure, approximately 59 hours were taken to finish the Iodine release and its removal from the RCS to achieve its target value for flood up. The original RF16 pre-outage plan had allotted 30 hours for RCS cleanup.

Chemistry monitored for the Cobalt Peak, which peaked at 5.04 uCi/ml on 10/23/2009 at 11:00. Following the peak, the last Reactor Coolant Pump was secured on 10/23/2009 at 12:05. Following RCS depressurization to atmospheric pressure, SDC purification was placed in service, which increased purification flow with CVC ion exchangers B & C in parallel.

Chemistry sample results determined that hard gamma was below the 0.05 uCi/ml flood up target value on 10/25/2009 at 13:26. This was determined by analyzing samples drawn in compliance with procedures.

RF16 dose rates on the SG primary side were elevated in comparison to RF15. Reasons for the elevation are as follows:

- The Reactor Trip caused a thermal hydraulic shock which resulted in a crud release. At the time of this release, the RCS was still in an alkaline reducing state. CRUD will plate out in the coldest part of the system; in this case, it was the Steam Generators (SG).

- The SG primary side was not drained as early during the outage as in some previous outages. The SG primary side was not drained until later in the outage (Day 17) to allow the SG nozzle dams to be installed with the reactor defueled rather than at hot leg mid-loop. This precluded the higher risk associated with installation of SG nozzle dams at reduced RCS inventory while SDC is needed to remove decay heat from the fuel in the reactor vessel. There is an increased risk to losing SDC at reduced inventory compounded with a reduced time to boil at the lower inventory. Avoidance of these risks, combined with a Co-58 peak of 5.0 uCi/ml, resulted in high activity water remaining in the Steam Generators.

At Waterford 3, Safety is continually reinforced and is tied to Nuclear, Personnel, and Radiological safety. Attention to ALARA was evident during RF16 in the decisions made to manage dose. Waterford 3 allocated 30 hours in the outage schedule for RCS cleanup in the pre-outage plan due to the anticipated abnormal radiological issues. This cleanup effort further demonstrates that Waterford 3 was aggressively including ALARA principles in its actions and was not being unjustifiably driven by outage schedule.

As an example, the following reflects abbreviated information from meeting minutes taken at a RF16 RCS DEI Clean-up Team during the outage that was discussing when to commence RCS pressure and temperature reduction to enter mode 5 versus RCS iodine activity levels and demonstrates some of the considerations being given to dose effects:

[BEGINNING OF ABBREVIATED MEETING MINUTES]

A RF16 RCS DEI Clean-up Team was established as an emergent issues team at 1540 on October 19, 2009. The team was established to determine if the original plan for holding RCS temperature and pressure at Mode 3 and not destaging the RCP seals until RCS DEI activity reached 0.0045 uCi/gm was still the right path given current conditions.

1. One option is to come down in RCS temperature and pressure all the way to Mode 5 without the currently scheduled hold for DEI activity and close containment access. If the containment activity is such that containment purge can not be effectively used for some time, this option could represent significant risk, taking an extended period of time to recover containment for personnel entry. There may be a point at which both containment purge and RCS cleanup can be effectively used together under this option – purging keeps up with the containment atmosphere activity when RCP seals are destaged. OE from Vogtle and Palisades was discussed. There were significant personnel radiological consequences at both plants. INPO has provided input that even 0.30 DAC (value from which the 0.0045 uCi/gm DEI target value is derived) may be too high from a down-stream radiological impact perspective. The team questioned offsite radiological effluent impact of not going all the way down and closing containment – the benefits/risks of this option vs. current hold plan from an effluents perspective. The team questioned the risks of changing from a plan that was well challenged and received much input to a different plan at this point in the outage. However, the team was open to change provided that any new plan or new plan elements will provide predictable results and will not result in significantly greater schedule impact or radiological consequence than the current plan. RCS clean-up using "feed and bleed" was discussed under this option. The team had received input at the time that using 3 charging pumps and ion exchange capability was just as effective/timely as "feed and bleed" without the added waste water volume consequence.

2. The other major option for consideration is to continue with the current plan to hold RCS temperature and pressure for DEI clean-up to 0.0045 uCi/gm. This plan minimizes radiological personnel consequences such as Iodine uptakes and hold ups at the RCA exits. This plan allows the use of 3 charging pumps for RCS clean-up in Mode 3. Entry to Mode 4 (under option 1 above) will require securing a charging pump and losing a third of the letdown flow for cleanup effectiveness. It may be possible to consider some modifications to this plan that achieve the desired result without implementing entirely the option in 1 above. This will be considered in the team's deliverable. Hydrogen peroxide addition to reduce Cobalt activity was also discussed. The team questioned at what RCS activity level hydrogen peroxide is added and whether or not it is the right point given elevated RCS Iodine activity for this outage and Iodine cleanup considerations. ANO OE on this item was discussed.

[END OF ABBREVIATED MEETING MINUTES]

The decision to continue on with the current plan was that the original plan was well thought out, had received several critical challenges, and had been incrementally improved based on the challenges. During the decision making process, input from INPO and EPRI was requested and there was ultimate consensus to continue with the established clean-up plan.

Waterford 3 controls outage schedule changes per procedure PLG-009-014, Conduct of Planned Outages, which provides guidance on the safe configuration and operation of the plant during outage execution. The procedure contains guidance that includes layers of reviews to determine the safety impact. For example, the Outage Risk Assessment Team evaluates individual work activities relative to potential impact to the key safety functions, including reactivity. Detailed reviews of the integrated schedule are conducted for correctness. This review includes an analysis for safety concerns. The ORAT performs a Risk Assessment based on plant condition windows and the recommended availability of key shutdown safety systems and the safety functions. The ORAT submits their review results to Onsite Safety Review Committee (OSRC) per the Pre-Outage Milestone Schedule that forwards the ORAT assessment to the GMPO for review and approval.

All revisions to the outage schedule are reviewed to determine if the change could affect the Shutdown Operations Protection Plan. A representative from Reactor Engineering or a currently licensed Senior Reactor Operator must review activities that may affect reactivity prior to their implementation. If approved revisions to the outage schedule constitute a Schedule Change, the Shutdown Operations Protection Plan shall be revised as necessary. The revised SOPP shall be reviewed and approved by an Assistant Outage Manager and ORAT prior to the work being performed. If a schedule change reduces defense in depth to a level that is below outage guidelines, a contingency plan shall be developed, when possible. The schedule change shall be reviewed and approved by an Assistant Outage Manager, the Outage Manager, or their designees, and ORAT.

Additional key decisions/actions:

Within two hours of the reactor trip, a proactive decision was made to expand the original DEI cleanup schedule to add 38 hours of RCS cleanup to the RF16 schedule.

After the plant was stabilized from the trip, purification flow was maximized by running three charging pumps.

In order to improve Iodine cleanup, the cooldown rate was reduced to 10 deg F/HR on 10/19/2009 at 23:22. This lowering of cooldown rate was performed to reduce the need for frequent lowering of letdown flow that would be needed to maintain pressurizer level within band. Lowering purification flow has the undesirable effect of reducing the Iodine cleanup rate.

The OCC logs for 10/21/2009 at 19:19 indicate that consideration was being given to the following emergent issue: when the RCS is depressurized to less than 400 psia, RCP seals were expected to destage and some amount of gas was expected to be released into containment. Engineering had provided this pressure threshold at pre-outage planning and was based on the pressure at which the seals would be destaged.

The Shutdown Cooling Trains were swapped to effectively feed and bleed Iodine from the RCS to the 600,000 gallon Refueling Water Storage Pool in order to reduce dose rates. This was decided because the amount of Iodine coming out of the fuel rods matched the capacity of purification and was keeping activity stable. This feed and bleed was the only available method at that time to lower Iodine activity. To perform the feed and bleed, high activity Reactor Coolant was established through one train of SDC, and then was swapped with the idle train that contained lower activity Refueling Water Storage Pool water. This water was then placed into the RCS to lower Iodine level. The now idle train was flushed of higher activity Reactor Coolant water to the Refueling Water Storage Pool.

On 10/23/2009 at 00:40, a decision was made to perform the forced liberation of hard gamma isotopes and further depressurization with Iodine levels greater than 0.025 uCi/ml. The logic was that a further depressurization to atmospheric was the ultimate goal in order to remove the last of the fission products from the fuel, through the defects, and into the RCS. After securing the two remaining reactor coolant pumps, this theory held true; RCS dose equivalent Iodine spiked from 0.02276 uCi/ml to 0.483 uCi/ml.

An additional factor considered for going to atmospheric pressure in the RCS was to reduce the ongoing leakage from the destaged RCP vapor seal. Depressurization of the RCS to atmospheric is also a prerequisite to place the Shutdown Cooling Purification in service. The Shutdown Cooling Purification loop provides a design flow of ~ 230 gpm versus the Chemical Volume Control letdown system which was only providing approximately 80 gpm. During Refuel 16 the 0.05 uCi/ml hard gamma flood up target value was reached in 50 hours. This included 10 hours of drain down where purification was secured. Use of the letdown system for cleanup would have required an additional 84 hours to reach the 0.05 uCi/ml hard gamma flood up target value.

ALARA Successes

- No radiological over-exposures to any worker, internal or external, including from radiography or diving operations.
- No posting errors.
- No shipping or control of Radioactive Material issues.
- No releases of airborne radioactivity from containment that exceeded current procedural limits.

Actions taken by Waterford 3 avoided 31 to 43 Rem of dose as discussed below:

Additional Shielding ~10-15 Rem avoided.

RCS Filtration ~3 Rem avoided: Installed two Tri-Nuc filters which reduced poolside dose rates to 3-5 mR/hr over a 10 day period.

Insulation Removal and Replacement ~ 10-15 Rem avoided: Removed additional blanket insulation from RCPs to reduce source-term. New insulation was reinstalled.

Other Actions ~ 8-10 Rem avoided:

- Frequent ALARA Committee meetings to look ahead for dose mitigation opportunities
- Purification flow improvements, SDC Flow Modifications, and SFP purification
- Shielding on the refueling machine bridge
- Behavioral changes and challenges – minimal crew sizes, more ALARA low dose areas

Section 7.0 RCP Vapor Seal Modification

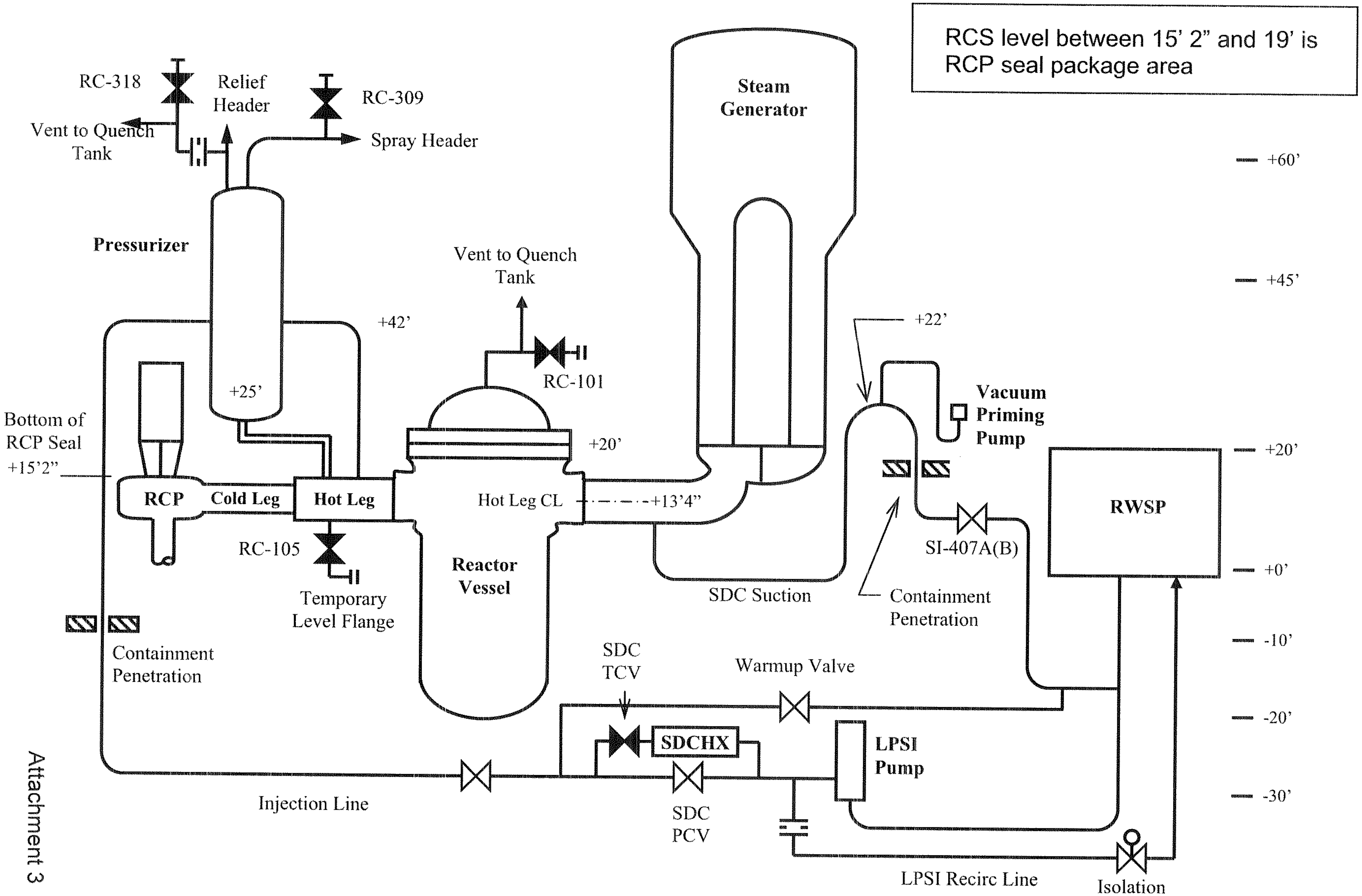
Waterford 3 has been working to install modifications to address the destaging of the N-9000 vapor seal. The N-9000 vapor stage modification was initially installed at WF3 during RF12 (Fall 2003). The first vapor stage seal de-staging issue was observed during the September 2005 shutdown for Hurricane Katrina. Actions to correct this condition were initiated with the vendor.

The design change from the vendor (Flowserve) was not complete in time for installation in RF15, and therefore not available for RF15 shutdown.

For RF16, there was an identified risk of installing the upgraded vapor stage at both W3 and ANO. Operating experience with the new design was limited. The decision to delay installation of the vapor stage seal modification supported obtaining ANO experience to validate the new seal performed as designed and did not create additional issues. Nonetheless, WF3 installed a modification in RF16 which captures the vapor stage leakoff and routes it to floor drains to reduce dose in future outages, mitigating the impact until the vapor stage seals are modified / upgraded.

Waterford 3's actions in this area have been consistent with ALARA principles.

FROM OP-001-003
RCS/SDC POSITION AND ELEVATION REFERENCE



Attachment 3

Response to Supplemental Questions

Introduction

Waterford 3 had submitted information in a white paper titled: White Paper Supplemental Communication of Possible White Violation for Radiation Protection Inspection. The lead inspector has requested additional information characterized as supplemental questions. The below information reflects the questions and Waterford 3's response.

It should be noted that the tables that explained the Radiation Work Permit (RWP) overages in our initial white paper implied that the at power reactor trip was a common reason that some of the RWPs were over original estimates. The wording generally stated: "Crud plate-out in Reactor Coolant System (RCS) piping from hard reactor trip (D-rings)" or "high RCS activity from hard reactor trip." Though the at power reactor trip created an initial crud burst that contributed to increased dose rates, there was also a high generation of hard gamma emitters from the hydrogen peroxide addition that affected the dose rates. The percentage of the dose from each occurrence is not easily ascertained; this is because the crud was also affected (solubulized to some extent) by the hydrogen peroxide addition.

Details

QUESTION RELATED TO SECTION 3

Please provide a copy of your pre-outage plan. Please provide ALARA Committee meeting minutes from the approval of outage dose estimates through the outage.

RESPONSE

Information provided in this package:

1) ALARA Committee Meeting Minutes from the approval of the outage dose estimates through the outage
2) Waterford 3, Refuel 16, Reactor Coolant System Dose Equivalent Iodine and Bulk Water Degassing Plan.
3) Meeting minutes taken at a RF16 RCS DEI Clean-up Team Meeting
4) Fragnet of pre-outage plan showing pre-planned clean-up actions
5) OSRC Meeting Minutes for 10/22/2009
6) Outage Risk Assessment Team report for RF16

Notable aspects of pre-outage plans

A cobalt peak of 3.0 uCi/cc from the H2O2 addition was predicted going into the outage. The RF16 schedule had securing the Reactor Coolant Pumps (RCPs) (1 in each loop running) 1 hour after confirmation of the cobalt peak and H2O2 residual. The schedule reflects:

Activity S/D00052 Perform H2O2 add to RCS	23-Oct 01:00
Activity S/D00125 Verify H2O2 residual	23-Oct 02:00
Activity S/D00086 Monitor for Cobalt 58 peak	23-Oct 02:00
Activity S/D00096 Secure RCP 2B & 1B	23-Oct 03:00

As indicated in our previous white paper, prior to Refuel 16, RCS Iodine-131 and Xenon-133 gas clean up activity levels were established. The goals were based on 0.3 DAC for Iodine and 0.03 DAC for Xenon, in the containment, with anticipated full destaging of the Reactor Coolant Pump Seals vapor stages at approximately 400 psi. This translates into an RCS Iodine-131 activity of 0.0045 uCi/ml and RCS Xenon-133 activity of 0.16 uCi/cc. The cleanup goals were captured in the Waterford 3, Refuel 16, Reactor Coolant System Dose Equivalent Iodine and RCS Degassing Plan. There were 30 hours placed in the Refueling Outage for reactor coolant system Iodine-131 and Xenon-133 cleanup.

Cobalt cleanup plans were developed which included a forced liberation of hard gamma isotopes at acid oxidizing conditions followed by running two reactor coolant pumps, one in each loop, until the Cobalt peaked. Once the peak was reached, plans included cleanup to the hard gamma target value of 0.05 uCi/ml prior to cavity flood up. The hard gamma activity target value of 0.05 uCi/ml was in place during previous outages and is in compliance with EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines. The hard gamma target value is also contained in procedure CE-002-006, Maintaining Reactor Coolant Chemistry.

The pre-outage plan directed that the plant would reduce pressure to between 1000 and 1200 psia to maintain control of letdown flow and provide margin to the destaging pressure of the RCP Seals and minimize vapor seal leakage.

Procedure CE-002-006, Rev 302, step10.9 USE OF HYDROGEN PEROXIDE (R) contains a note stating:

Addition of hydrogen peroxide with Reactor Coolant Pump running in each loop is the preferred method for hydrogen peroxide addition. Deviation from this method may have the following consequences due to inadequate mixing in Reactor Coolant Loops and Steam Generators:
- *Reactor Coolant Loops potentially remain hydrogenated with insufficient dissolution of nickel.*
- *Upon opening of Steam Generator primary side manways, Iodate formation may result due to rapid oxygenation.*

Plant management should be made aware of potential dose considerations from Reactor Coolant Loops.

Summary of outage execution challenges:

On October 19, 2009 at 0945 hours, Refuel 16 was started early when the reactor was manually tripped from approximately 100% reactor power. This caused a thermal hydraulic shock (at power trip) which resulted in a CRUD release. At the time of this release, the RCS was still in an alkaline reducing state. CRUD plated-out in the coldest part of the Reactor Coolant System (RCS): the Steam Generators (SG).

At Power Reactor Trip Response (Information obtained from interviews with decision makers)

It was considered that the at-power trip would impact the crud burst. Thoughts were that any efforts to clean up iodines would result in hard gamma clean up as well, especially after Chemistry solubulized the particulate contaminants. It was believed that mitigating the most limiting concern at the time (Iodines) also encompassed hard gamma cleanup as well.

Co-58 (hard gamma) chemistry data at that point in time was as expected. Our success path was to depressurize the RCS to atmospheric and place Shutdown Cooling (SDC) purification in service. This action would maximize our purification flow and provide the best cleanup of the RCS and reduction in dose rates.

EPRI guidance wants as much time as available in the acid-reducing phase because of its benefit to reduce outage dose. The acid-reducing phase occurs at the beginning of an outage once the RCS is borated; Waterford 3 was in this condition from 10/19/09 through 10/23/09.

Decision to Depressurize (Information obtained from interviews with decision makers)

We were cleaning for many hours and the clean-up rate had reached its plateau. Every time we depressurized, more came out, the clean-up rate picked up, then reached another plateau. We got to the point where we needed secure the RCPs and depressurize to achieve final clean-up. We felt that moving swiftly through depressurization, getting all the fission products out of the fuel pins, and then putting SDC purification in service was the best way to clean up; we can almost triple the purification flow through the CVC ion exchangers using the SDC purification system. We were focused on time to clean-up and rate, because time is dose in an outage. We also were focused on depressurization to stop the leakage in the D Rings from the destaged RCP vapor seals.

Getting onto shutdown cooling and securing the RCP's (Information obtained from interviews with decision makers)

A significant concern that was being dealt with was the iodine and xenon gas being released by the failed fuel into the RCS. The levels were not meeting specifications and the out-gassing was equalizing with the removal capability, causing an equilibrium value that was unacceptable for securing cleanup. This issue had not been acceptably dealt with at other sites and caused iodine uptakes and loss of containment due to gas levels. Because of this Operating Experience (OE), this was a major focus.

In RF16 we experienced RCS activity higher than anticipated. It was recognized by the OCC that RCS cleanup would have to be extended in order to reduce this activity. This cleanup was our focus during RF16 when we were preparing the plant for maintenance.

The iodine levels had stabilized above the acceptance criteria. Every time pressure in the RCS was lowered, more iodine was released. It was necessary to reduce pressure both to cleanup the iodine and to get to pressure and temperature limits where hydrogen peroxide could be added. This meant going on to shutdown cooling in parallel with two RCP's running.

At the pressure when hydrogen peroxide was added, cleanup flow through the letdown heat exchangers was approximately 85 GPM. The cleanup rate would not achieve the cleanup target because we had not yet fully depressurized and would take an excessively long time to clean up the cobalt. A new modification provided a 235 GPM cleanup rate with shutdown purification in service for cobalt cleanup. The much shorter time for cleanup was a prime consideration to get the dose rates down as quickly as possible. This turned out to be successful.

RCP seal leakage was causing high dose rates in the RCP insulation and high contamination levels in the D-rings. There were concerns about PCEs and potential uptakes; depressurizing the RCS reduces the seal leakage and would mitigate this problem.

<u>Install nozzle dams prior to defueling</u> (Information obtained from interviews with decision makers)

There was a common mindset that this was not a good idea. Going to reduced inventory/mid-loop to install the nozzle dams is a significant challenge to nuclear safety that we will avoid when possible. During RF16, it was possible to avoid reduced inventory and get the core offloaded without the additional risk of reduced inventory operation. Plant Risk was maintained GREEN during the entire outage as an outcome.

Discussion were held in the Outage Support Center, but because of this common understanding, a formal challenge was not held during the outage on altering the schedule to emerge a reduced inventory earlier in the schedule. Additionally, the schedule complexity (with thimble tube replacement, RCP pump replacement, SG inspection, alloy 600 cold leg and water inventory plan, the challenges of expediting the nozzle dam installation task along with the attendant supporting work) was considered too major and risky a change to accommodate an early draining of the RCS.

<u>Hydrogen Peroxide Addition (after securing the RCP rather than while the RCP's were running)</u> (Information obtained from interviews with decision makers)

Adding hydrogen peroxide after securing the RCPs was a major concern by Chemistry. This was not a normal means of peroxide injection and it was uncertain if there would be unintended consequences. This method was not pre-planned for this site

Procedure CE-002-006, which addresses hydrogen peroxide addition, was changed during the outage with an effective date of 10/22/09. The OSRC meeting minutes for this procedure (also held on 10/22/09, which followed the at-power trip) reflect that curd burst and cobalt activity were discussed in great detail. The minutes also indicate that discussion included: "inadequate mixing without the reactor coolant pumps will cause a lingering affect." Crud release would occur when system was opened up and would be uncontrolled.

We wanted the RCPs running to get full circulation of the peroxide. There was also a concern about opening the system and an Iodine/gas release. It was recalled that the decision from OSRC was that it was doable, but not recommended as a first choice.

On October 23, 2009, after extensive efforts of Iodine mitigation in the RCS, hydrogen peroxide was added to the RCS to conduct a forced liberation of Cobalt (hard gamma-emitting isotopes).

On October 24, 2009 (following the H2O2 addition), dose rates on the #1 RCS loop were observed to be approximately 2.22 times higher than pre-planned dose rates, up to 100 mR/hr on the Steam Generator #1 hot leg and cold leg platforms (Reference Condition Report CR-WF3-2009-05886). Waterford 3 Management and Radiation Protection mitigated dose on the #1 RCS loop by increasing the amount of temporary shielding and curtailing work for some groups until dose rates were near pre-planned values.

Additional discussion

Throughout the outage, multiple discussions were held at the fleet and site management levels on the radiological conditions current at the time with respect to affects on the outage dose and schedule. Decisions were made based on RCS activity, potential clean-up methods, dose mitigation activities, and expected impacts on both the RWP dose and outage duration. There where overriding factors considered of no iodine uptakes to personnel and no automatic trip of Containment purge, which could have led to exceeding gas effluent limits.

On October 20, 2009 at ~ 0025 hours, initial entries were made into the Reactor Containment Building (RCB) to perform initial habitability surveys. Based on radiological survey data, it was determined that both radiation levels and contamination levels were within historical values in most areas of the RCB, while levels on the SG platforms and at the RCPs were slightly elevated. This can be seen by comparing the RF-15 to RF-16 radiological surveys provided separately.

There were several variances in the plant shutdown plan from previous outages. The SG primary side was not drained in the usual time frame. Hydrogen peroxide (H2O2) is usually added within 24 hours and Steam Generator primary side is drained by 4th day of outage. In RF-16, peroxide was not added until 4th day of outage. The delay in adding peroxide was a result of trying to meet an Iodine activity target prior to the addition. ANO had previous OE on Iodate being formed when Iodine levels were above the target of 4.5E-3 uCi/ml. Iodate is an insoluble form of Iodine that cannot be removed by Ion Exchange. With the RCS in an acid reducing state for this extended period of time (4 times longer) more CRUD was reduced (burst) from the entire RCS and made soluble than in previous outages. In addition, this high activity water was trapped in the Steam Generator U-tubes for 13 days and allowed settling and further plate out of CRUD in the Steam Generators. Though the at power reactor trip created an initial crud burst that contributed to some of the increased dose rates, there was also a high generation of hard gamma emitters from the hydrogen peroxide addition that affected the dose rates. The crud was also affected (solubulized to some extent) by the hydrogen peroxide addition.

QUESTIONS RELATED TO SECTION 4

You state, "That is, leakage of high activity water from the RCP seals cannot be avoided." Knowing this, what actions were planned to mitigate the effects of highly contaminated water released to work areas? Please highlight the discussion, if these actions are included in the pre-outage plan. Has similar leakage occurred previously?

Which reactor coolant pumps leaked significantly when RCS pressure dropped below 400 psia? (The apparent cause evaluation in CR-WF3-2009-07262 states, "During RCS depressurization to reach Mode 5 and at approximately 350 psia, as anticipated, all four reactor coolant pump seal vapor stages de-staged and leaked highly contaminated RCS water onto pump insulation, adjacent structures, and to -11 RCB." Other references, such as log entries seem to indicate that all reactor coolant pumps did not leak significantly. The apparent cause evaluation in CR-WF3-2010-0990 states, RCP 1B and RCP 2B insulation was a significant source of radiation and was removed emergently during RF-16.") Please provide surveys of each reactor coolant pump and surrounding areas from shutdown through the next 14 days.

The RCP seals were redesigned to correct the leakage problem. When were the seals installed at ANO?

RESPONSE

The surveys of each reactor coolant pump and surrounding areas from shutdown through the next 14 days are provided along with this response.

What actions were planned to mitigate the effects of highly contaminated water released to work areas? Please highlight the discussion, if these actions are included in the pre-outage plan.

The actions planned to mitigate the effects of highly contaminated water released to work areas were to follow sound Radiation Protection (RP) practices for granting access to radiation worker. This included initial containment entries to survey areas requiring access and establishing the proper controls and restrictions for worker entry so that mitigation of dose could be achieved. "Go" criteria was pre-established for work in the D-rings and RCPs, which is reflected in file "D-Ring_Go Criteria.pdf" (provided separately).

These actions are embedded throughout the D-Ring_Go Criteria.pdf and are highlighted in the Waterford 3, Refuel 16, Reactor Coolant System Dose Equivalent Iodine and Bulk Water Degassing Plan (also provided separately).

Similar to RF15, RF16 pre-planned dose reduction/contamination reduction plans included the following actions to mitigate the effects of highly contaminated water released to work areas:

Steam Generator "Go" criteria was pre-established based on dose rates for Steam Generator #1 and Steam Generator #2 to support RWP dose estimate planning.

There was planned decontamination of D-rings to mitigate expected Reactor Coolant Pump Seal Leakage.

Pre-planned cleaning of Reactor Containment Building drains on -11 elevation to ensure any water leakage from RCP seal packages would flow freely to the RCB sump. This activity is performed by the RP group prior to the RCP seals de-staging.

Installation of temporary shielding on the Safety Injection lines to lower dose rates to personnel working on the Steam Generator platforms.

Has similar leakage occurred previously?

The N-9000 vapor stage modification was initially installed at WF3 during RF-12 (Fall 2003). The first vapor stage seal de-staging issue was observed during the September 2005 shutdown for Hurricane Katrina, which followed RF-13. Actions to correct this condition were initiated with the vendor.

During RF-14 in November 2006, CR-WF3-2006-03597 documented that RCP-2B Vapor Stage opened during plant shutdown.

During RF-15, CRs CR-WF3-2008-1678, -1679, -1682, and -1683 document that during boric acid walk downs after plant shutdown, evidence of leakage was noted from each of the RCPs seals

Which reactor coolant pumps leaked significantly when RCS pressure dropped below 400 psia?

All 4 RCPs are currently susceptible to vapor stage de-staging and all 4 RCPs destaged and leaked in RF16.

On 10/22/09, the RCP system engineer performed an initial walk down of the RCPs for seal destaging. It was noted that RCP 2B and 1A vapor stages were leaking ~0.25-0.50 GPM. RCP 1B was not walked down; however, the leakoff line at -11 elevation appeared to have ~0.25-0.50 GPM coming out of it. RCP 2A did not appear to have any visible leakage. Leakage from RCP 2A was first recognized on 10/25/09 during RCS depressurization to reach Mode 5 and at approximately 350 psia. This was concurrent with the increased leakage from the other 3 RCPs as documented in CR-WF3-2009-07262.

The crud burst from the hydrogen peroxide addition peaked on 10/23/09 and the last two RCPs were secured at that time. On 10/31/09, it was noted that when Reactor Coolant System leaked from the RCP 1B and 2B seals, RCS radioactivity was at higher concentrations due to that crud burst.
An additional factor in the varying contamination levels was that RCS loop 1 (containing SG #1) had SDC in operation in that loop for only ~ 3 hours out of the 5 days that Waterford 3 was on SDC prior to flood up of the reactor cavity. With this minimal opportunity to dilute loop 1, the leakage from RCS loop 1's RCPs would have a higher prospect for introduction of contaminates than in RCS loop 2.

When were the seals installed at ANO?

ANO Unit 2 has installed the Vapor Stage Spring Modification on two of their Reactor Coolant Pumps during their 2009 Refueling Outage and is planning on installing the remaining two during their 2011 Refueling outage.

QUESTION RELATED TO SECTION 5

You state the target goals for reactor coolant system cleanup were met. The apparent cause evaluation in CR-WF3-2009-07262 states, "When the refueling cavity was filled on October 26, 2009, dose rates near the refueling cavity were a factor of 3 to 5 times higher than planned to support the refueling radiation work permits. Radiation work permits for refuel work (700 series) were planned using an average dose rate of 3 mR/hr around the perimeter of the refueling cavity. Contrary to this, dose rates around the refuel cavity currently range from 8 mR/hr to 15 mR/hr." If you reached your target values for cleanup, to what do you attribute the higher dose rate?

What were the activity levels in the steam generators versus the chemistry target values? (The apparent cause evaluation in CR-WF3-2009-07262 references a cobalt-58 peak of 5 microcuries/milliliter.)

RESPONSE

To what do you attribute the higher dose rate?

The higher dose rates around the refuel cavity are attributed to elevated activity in the Refueling Water Storage Pool resulting from RCS clean-up actions and the eventual purging of the remaining activity in the Steam Generator U-tubes to the cavity during flood-up.

The Shutdown Cooling Trains were swapped to effectively feed and bleed Iodine from the RCS to the 600,000 gallon Refueling Water Storage Pool in order to reduce iodine levels to prevent source gamma issues. The amount of Iodine coming out of the fuel rods matched the capacity of purification and was keeping activity stable and unacceptably high. This feed and bleed was the only available method at that time to lower Iodine activity. To perform the feed and bleed, Reactor Coolant was established through one train of SDC, and then was swapped with the idle train that contained lower activity Refueling Water Storage Pool water. This water was then placed into the RCS to lower Iodine level. The now idle train was flushed of higher activity Reactor Coolant water to the Refueling Water Storage Pool.

Following this action, the Refueling Water Storage Pool was cleaned to the following specifications prior to use of the water for flooding the refueling cavity, based on data recorded for 10/25/2009 08:55.

I-131	0.000957	uCi/ml
Gross Activity	0.0386	uCi/ml
Hard Gamma	0.00560	uCi/ml

Since these values were higher than pre-outage levels, there was a reduction in the dilution effect when the RCS and RWSP were blended upon flood-up. Gross Activity in the RWSP prior to the feed and bleed was 0.000273 uCi/ml as measured on 9/22/09.

On October 27, 2009, the B Train LPSI pump was used to fill the cavity, which we believe purged the remaining activity in the Steam Generator U-tubes to the cavity during flood-up. This also contributed to elevating the refueling cavity activity.

What were the activity levels in the steam generators versus the chemistry target values?

The activity levels on the primary side of each steam generator is not known. Each steam generator was subject to differing dynamics during that period that cannot be easily ascertained without making some unsubstantiated assumptions. Waterford 3 is unable to obtain a sample in that region of the RCS.

It is assumed, however, that the initial activity levels in the steam generators were the same as RCS activity at the time the last RCP was secured. Cobalt peaked at 5.05 uCi/ml and the last Reactor Coolant Pump was secured on 10/23/2009 at 12:05. It is assumed that primary side of SG #2 in loop 2 was progressively flushed by having Shutdown Cooling Loop A in operation for an extended period, and its activity level was relatively tracking near the RCS activity level. It is also assumed that SG #1 in loop 1 was subject to some dilution as well from diffusion between the concentrations existing in the system. Nonetheless, the activity level in SG #1 cannot be as readily ascertained.

Chemistry goals for RCS cleanup are located in CE-002-006, Maintaining Reactor Coolant Chemistry lists the recommended target values for RCS cleanup in Section 10.8 as follows:

- *I-131 < 0.01 uCi/ml*
- *Xe-133 < 0.5 uCi/cc*
- *Hard Gamma Emitters (Co-58 + Co-60 + Cs-134 + Cs-137 + Mn-54) < 0.05 uCi/ml*

QUESTION RELATED TO SECTION 6

You state, "The steam generator primary side was not drained as early during the outage as in some previous outages. The steam generator primary side was not drained until later in the outage (Day 17) to allow the steam generator nozzle dams to be installed with the reactor fueled, rather than at hot leg mid-loop." Was consideration given to draining or flushing the steam generators to remove the highly contaminated water? Was this possible for your steam generators? Was operating experience related the removal of source term from the steam generators at other sites reviewed? If so, is your review and consideration documented (such as in ALARA committee minutes)? Please provide a copy. Please provide representative area surveys around the steam generators from shutdown through the next 21 days.

RESPONSE

Representative area surveys around the steam generators from shutdown through the next 21 days are provided along with this response.

Was consideration given to draining ... the steam generators to remove the highly contaminated water?

Draining the steam generators to remove the highly contaminated water was not considered a reasonable option because that would have required us to install the nozzle dams prior to defueling. Going to reduced inventory/mid-loop to install the nozzle dams is a significant challenge to nuclear safety that we will avoid when possible. During RF16, it was possible to avoid reduced inventory and get the core offloaded without the additional risk of reduced inventory operation. Plant Risk was maintained GREEN during the entire outage as an outcome.

Discussions were held in the Outage Control Center concerning this option, but because of this common understanding, a formal challenge was not held during the outage on altering the schedule to do this. Additionally, the schedule complexity (with thimble tube replacement, RCP pump replacement, SG inspection, alloy 600 cold leg and water inventory plan along with the challenges of expediting the nozzle dam installation task along with the attendant supporting work) was considered too major and risky a change to accommodate an early draining of the RCS.

Was consideration given to ... flushing the steam generators to remove the highly contaminated water?

Flushing the steam generators to remove the highly contaminated water was not considered a reasonable action because this would have required us to continue to operate the RCPs for an extended period. As discussed above, a significant concern that was being dealt with was the iodine and xenon gas being released by the failed fuel into the RCS. The levels were not meeting specifications and the out-gassing was equalizing with the removal capability, causing an equilibrium value that was unacceptable for securing cleanup. This issue had not been acceptably dealt with at other sites and caused iodine uptakes and loss of containment due to gas levels. Because of this OE, this was a major focus.

It was recognized by the OCC that RCS cleanup would have to be extended in order to reduce this activity. The iodine gas levels had stabilized above the acceptance criteria. Every time pressure in the RCS was lowered, more iodine was released. It was necessary to reduce pressure both to cleanup the iodine and to get to pressure and temperature limits where hydrogen peroxide could be added. This meant going on to shutdown cooling in parallel with two RCP's running.

At the pressure when hydrogen peroxide was added, cleanup flow through the letdown heat exchangers was approximately 85 GPM. The cleanup rate would not achieve the cleanup target because we had not yet fully depressurized and would take an excessively long time to clean up the cobalt. A new modification provided a 235 GPM cleanup rate with shutdown purification in service for cobalt cleanup. The much shorter time for cleanup was a prime consideration to get the dose rates down as quickly as possible.

RCP seal leakage was causing high dose rates in the RCP insulation and high contamination levels in the D-rings. There were concerns about PCEs and potential uptakes; depressurizing the RCS reduces the seal leakage and would mitigate this problem. Stopping this leakage required us to depressurize which also required securing the RCPs.

Was draining or flushing the steam generators to remove the highly contaminated water possible for your steam generators?

Though it is possible to drain or flush our steam generators to remove the contaminated water by nozzle dam installation or maintaining RCP operation for an extended, this was not performed for the reasons stated above.

Was operating experience related the removal of source term from the steam generators at other sites reviewed? If so, is your review and consideration documented (such as in ALARA committee minutes)? Please provide a copy.

During pre-outage planning, dose mitigation plans had been developed that included RCS clean-up actions. RCS clean-up practices used at Waterford 3 had been historically effective and there was no indication during pre-planning that would we should search for specific OE focusing on the removal of source term from the steam generators. Though no individual site's operating experience related to the removal of source term from the steam generators was reviewed, the EPRI document is a compilation of OE associated with RCS clean-up going into an outage. For example, the EPRI document includes discussion on the impact of post-flood-up releases on fuel pool Co-58 concentrations at the Byron and Braidwood nuclear plants.

As indicated in our previous white paper, the EPRI guideline allows securing of the Reactor Coolant Pumps (RCPs) following the addition of hydrogen peroxide followed by RCS cleanup to target values via the shutdown cooling purification system. The sequence followed during this outage up to placing SDC purification in service, including the timing of securing the RCPs following the Cobalt peak, has been the approach taken in the past. This methodology follows the guidance in the EPRI Pressurized Water Reactor Primary Water Chemistry Guideline.

Additionally, Waterford 3 follows Entergy fleet procedures that screen OE for its impact on the site; follow-up reviews and actions are taken where determined warranted. If there had been OE on the removal of source term from the steam generators, it would have been reviewed for any additional insights it may have offered beyond the OE contained in the EPRI document.

References to OE usage during pre-RF-16 outage planning

Source: Failed Fuel Procedure – Fort Calhoun
Used In: RP Preps for an Outage Following Failed Fuel.doc
Topic: mitigating outage dose with failed fuel

Source: Palisades OE
Used In: W3 Outage Readiness for Failed Fuel Recommendations.doc
Topic: mitigating outage dose with failed fuel

Source: EPRI shutdown guidelines
Used In: W3 Outage Readiness for Failed Fuel Recommendations.doc
Topic: chemistry control to mitigate outage dose

Source: INPO Assist Visit, May 2009
Used In: entire RF-16 planning
Topics:

- Importance of all personnel realizing the impact of RCS leakage when RCP seals destage during planning activities.
- Containment Purge flow must be maximized to insure discharge of gases via filtered stack vice maintenance hatch.
- All fuel returning to the reactor must be sipped. Two other PWRs have reloaded damaged fuel assemblies during the last two years due to ineffective fuel sipping.
- Effect of high Iodine spike at shutdown must be accounted for in planning and evaluated against potential E-Plan and Technical Specification challenges.
- Track planning actions to reduce CRE in single tracking tool to ensure site cohesion.
- Ensure ALARA Committee review for all major projects.

Additionally, there was ongoing communication with EPRI to obtain their insights on best industry practices and industry operating experiences.

Waterford 3 Response to NRC Questions

Question 1:

In your response to supplemental questions, you stated, "The first vapor stage seal de-staging issue was observed during the September 2005 shutdown for Hurricane Katrina, which followed RF-13. Actions to correct this condition were initiated with the vendor. During RF-14 in November 2006, CR-WF3-2006-03597 documented that RCP-2B Vapor Stage opened during plant shutdown. During RF-15, CRs CR-WF3-2008-1678, -1679, -1682, and -1683 document that during boric acid walk downs after plant shutdown, evidence of leakage was noted from each of the RCPs seals.

The NRC asked, "What actions were planned to mitigate the effects of highly contaminated water released to work areas?" You responded, "The actions planned to mitigate the effects of highly contaminated water released to work areas were to follow sound Radiation Protection (RP) practices for granting access to radiation worker."

There was no discussion of catch basins, drain lines, or other engineered control to route the contaminated water away from work areas. Were such measures implemented? If so, when were they implemented and what was the effect? Was Refueling Outage 16 the first time vapor seal leakage flowed onto pump insulation or adjacent structures or to -11 foot elevation of the reactor containment building?

Response to question 1:

Waterford 3 has installed drain line modifications and temporary Leakage Collection Devices to address the destaging of the N-9000 vapor seal. In October 2007, CR-WF3-2007-03716 documented that Reactor Coolant Pump vapor stage leakage was caused by vapor stage quad ring hanging up and that the vapor stage leakoff line was not draining to the Reactor Drain Tank. As a result in RF-15 (April 2008) Vapor Stage leakoff lines were rerouted to a floor drain rather than the Reactor Drain Tank (installed under EC 6256). Nonetheless in May 2009 (RF-16) CR-WF3-2009-5501 documented that Boric Acid was discovered in RCP 2B partially due to heat exchanger gasket leakage, vapor stage leakoff line not performing its design function, and quad ring hang up. The root cause of inadequate design of the vapor stage seal leak-off line was caused by the check valves installed on the leak-off lines were incapable of passing flow as intended by design. EC 18520 was implemented on all four Reactor Coolant Pumps to address this issue in May 2009 (RF-16) to reroute the vapor stage leakoff line from each pump to individual floor drains, by-passing the in line check valve.

The temporary leakage collection devices described below refers to drain plugs and drain lines installed in the Reactor Coolant Pump shroud areas. The timeline for installation of leakage collection devices to divert leaking Reactor Coolant System from the Reactor Coolant Pump (RCP) vapor stage seals is as follows:

Drain plugs and drain lines for RCP 1A were installed in the beginning of Refuel 16 (prior to hydrogen peroxide addition to the Reactor Coolant System). As a result, there was no leakage on the RCP insulation package or the area below.

Drain plugs and drain lines for RCPs 1B, 2A and 2B were started at the beginning of Refuel 16 (prior to hydrogen peroxide addition to the Reactor Coolant System); however, they were not completed due to high dose rates being created from the leaking RCS from the vapor stage seals. The drain plugs and drain lines for RCPs 1B, 2A and 2B were subsequently installed after insulation blankets were removed and area decontamination was performed. The completion of leakage collection devices on RCP 1B, 2A and 2B was on November 5, 2009.

The leakage collection devices for RCPs 1B, 2A and 2B were not as effective as performed during Refuel 15 because all of the drain plugs and drain lines were not installed due to high dose rates (up to 5 R/hr) created from the leaking RCS through the RCP vapor stage seals.

Installation of Leakage collection devices were performed in the same manner during Refuel 16 as was performed in Refuel 15. The delta between Refuel 15 and Refuel 16 in the installation of leakage collection devices is attributed to the higher than anticipated leaking Reactor Coolant System water and the higher than anticipated dose rates attributed to the 5 microcurie/ml leakage from Reactor Coolant Pump vapor stage seals.

Refuel 16 was not the first time that Reactor Coolant System leakage flowed onto the pump insulation or adjacent structures to -11 foot elevation of the Reactor Containment Building. This condition occurred during Refuel 15 also. While this condition occurred during Refuel 15, the effect was not as significant due to the lesser amount of RCS leakage and the radioactivity was higher in Refuel 16 as compared to Refuel 15 (3 microcuries peak Cobalt in Refuel 15 compared to 5 microcuries in Refuel 16).

Question 2:

Per your responses to the NRC supplemental questions, you have assumed some dilution within the steam generator loops, but have no data that shows that these activity levels met targeted values for RCS cleanup in accordance with Procedure CE-002-006 (as defined by EPRI guidance). You state, "Waterford 3 is unable to obtain a sample in that region of the RCS" and "... the activity level in SG #1 cannot be as readily ascertained."

How can you assure that you have reached the targeted chemistry goals for RCS cleanup, if you cannot or did not accurately ascertain the level of activity in the steam generator loops, which is inclusive to the RCS? If sampling was available in that region of the RCS, why was it not completed?

Is Procedure CE-003-327, "Operation of the Primary Sample Panel," Step 10.1, "Obtaining Depressurized RCS Hot Leg Sample" applicable to sampling water in the steam generators? Does Training Material Number SD-PSL describe a sampling point (P1 or others) from which of water from the steam generators could have been sampled?

Response to question 2:

Waterford 3 has acknowledged that the activity level in the steam generator loops cannot be readily ascertained. Nonetheless, in Waterford 3's response to the NRC's supplemental questions, Waterford 3 indicated that the targeted chemistry goals for RCS cleanup were obtained in compliance with the requirements of procedure CE-002-006, Maintaining Reactor Coolant Chemistry, Section 10.8 "Reactor Coolant System Chemistry Control for Refueling." Specifically, the RCS cleanup requirements of CE-002-006, Section 10.8, Step "m".

The only way a pressurized water reactor designed plant can assure the Steam Generator U-tubes (loops) are cleaned is if the RCPs are maintained in operation. However, as indicated in a previous response, CE-002-006 Section 10.8 and the EPRI guideline both allow securing of the Reactor Coolant Pumps (RCPs) following the addition of hydrogen peroxide followed by RCS cleanup to target values via the shutdown cooling purification system.

Waterford 3 procedure CE-002-006 (Maintaining Reactor Coolant Chemistry) follows the EPRI-1014986 (Pressurized Water Reactor Primary Water Chemistry Guidelines) guidance. The NRC perspective that the Steam Generator U-tubes inventory must be included in the chemistry target value differs from the EPRI-1014986 guidance and Waterford 3 procedure CE-002-006. The NRC perspective would be an industry issue that should be addressed in the EPRI-1014986 guidance because the EPRI information is the basis for the Waterford 3 CE-002-006 procedure.

The sequence followed during this outage up to placing SDC purification in service, including the timing of securing the RCPs following the Cobalt peak, has been the approach taken in the past. This methodology follows the guidance in the EPRI Pressurized Water Reactor Primary Water Chemistry Guideline.

Sampling is not available in the Steam Generator U-tube region of the RCS. Procedure CE-003-327, "Operation of the Primary Sample Panel," Step 10.1, "Obtaining Depressurized RCS Hot Leg Sample" does not provide for sampling water directly on the primary side of the steam generators. Training Material Number SD-PSL does not describe a sampling point (P1 or others) from which water from the steam generators could have been directly sampled.

Waterford 3 Response to NRC Questions

RCS Sampling Capabilities Description (supporting response to Question 1)

The SDC System uses the LPSI system for its flowpath, utilizing the LPSI Pump for the motive force for circulation. During shutdown cooling operation, a portion of the RCS is diverted to the SDC System headers via the SDC nozzles located in the RCS hot legs and directed through the two SDCHXs. The SDCHX is realigned to the LPSI system from CS and the LPSI Pump minimum flow re-circulating valves are closed to prevent draining the RCS into the Refueling Water Storage Pool (RWSP). Reactor coolant is circulated by the LPSI Pump through the SDCHX to the RCS cold legs, via the LPSI header, through the safety injection nozzles. When the SDC System is operational, a flow path through the Chemical and Volume Control System (CVCS) can be established to remove fission and activated products. This is accomplished by diverting a portion of the flow from the SDCHX discharge through SI-418A(B) and SI-423 to the letdown line upstream of the Letdown Heat Exchanger.

The following points are available for sampling the Reactor Coolant System.

- P1 RCS hot leg #1
- P2 Pressurizer (PZR) surge line
- P3 Pressurizer steam space
- P4A Shutdown cooling suction line "A"
- P4B Shutdown cooling suction line "B"
- P5A HPSI pump "A" miniflow line
- P5B HPSI pump "B" miniflow line
- P6 Purification filter inlet
- P7 Purification filter outlet
- P8 Ion exchanger outlet
- P9 Volume Control Tank (VCT)

Shutdown cooling taps into the RCS hot and cold legs where flow is directed through the reactor vessel and not through the Steam Generators. Since the sample points listed above fall within the Shutdown Cooling flow path, none of the sample locations would have provided sampling of the Steam Generator primary side contents in the condition the plant was in.

Historically the RCS hot leg or the Purification filter inlet has been used for sampling of RCS to determine if the 0.05 uCi/ml flood up limit has been reached. This protocol was used in refuel 16.

From: POLLOCK, JIM [mailto:RPOLLOC@entergy.com]
Sent: Thursday, March 17, 2011 1:33 PM
To: Ricketson, Larry
Cc: STEELMAN, WILLIAM J
Subject: Waterford 3 RCP Leakage Flow Experience

Larry,

You had called for clarification on vapor seal leakage flowing onto pump insulation or adjacent structures or to -11 foot elevation of the reactor containment building. Please review the below information and let me know if it is what you needed.

During Hurricane Katrina (September, 2005), it was noted that there was leakage in Reactor Coolant Pump 1B and 2B seal housing areas. This was noted by Engineering to be attributed to the RCP vapor stage seal faces opening allowing leakage through the seal flange area. Condition Report CR-WF3-2005-3831 noted this leakage from RCPs 1B and 2B. The seal leakage flow path made it to the -11 foot elevation of the reactor containment building. In order for the leakage from the seals to get to the -11 foot elevation at ~ 3 to 5 GPM, leakage out of the immediate seal area onto pump insulation would most likely have had to occur.

During Refuel 14, (November, 2006), RCP 1B and 2B leaked onto the respective RCP insulation packages. During Refuel 14 leakage from the Reactor Coolant pumps onto the insulation under the shrouds raised the effective dose rate for Reactor Coolant pump work 13%-35% from the as left effective dose rates for similar work performed during Refuel 13. RF-14 had a 3.18 microcurie Cobalt 58 peak

During the 2007 Mid-Cycle Outage (October, 2007), RCPs 1B and 2B leaked onto the respective RCP insulation packages as indicated by radiological survey data.

During Refuel 15, (April 2008), leaking Reactor Coolant System water eventually leaked past the installed drain plugs and drain lines on Reactor Coolant Pumps. While the RCS leaked onto some of the insulation, the impact from a dose standpoint was not to the magnitude as noted in Refuel 16. This was primarily due to the drain plugs and drain lines being installed earlier in the outage because the dose rates from the leaking RCS was small (3 microcurie Cobalt 58 during Refuel 15 versus 5 microcurie Cobalt 58 peak during Refuel 16). The resulting leakage in Refuel 15 was not a significant issue during Refuel 15 because the work scope on Reactor Coolant Pumps during the leakage time frame was smaller in comparison to Refuel 16 where Reactor Coolant Pump 1A scope had a complete pump and motor replacement (more person hours spent).

Jim Pollock

8-580-6561

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Waterford 3 had previously provided an initial response titled "WHITE PAPER PRELIMINARY COMMUNICATION OF POSSIBLE WHITE VIOLATION FOR RADIATION PROTECTION INSPECTION." Clarification was requested of the information contained in section 6.0 ALARA Decisions and Actions for Work Activities, which presented a summary table illustrating the dose impact on each RWP from associated job activities. The requested clarification is discussed below.

1) Please include RWP 20090617 in the discussion using the same format presenting the other RWPs.

See page 3 for information.

2) In section 6.2, RWP 20090513 indicates initial dose estimate of 13.590 REM; in conflict with this an initial dose estimate of 12.775 REM which was reflected in the Refuel Outage 16 Radiation Protection Report provided before the inspection. The 13.590 appears to be the dose associated with Revision 1. Please clarify the actual initial dose estimate.

RWP 2009-0513 had an actual initial dose estimate of 13.590 person-rem, as documented in Rev. 0 of RWP 2009-0513, which was signed by the ALARA Committee Chairperson on 8/3/2009. RF16 began on 10/19/2009. The Refuel Outage 16 Radiation Protection Report reflected 12.775 REM in error.

3) Some of the RWPs evaluated in section 6.2 do not account for the total dose difference. For example, RWP 20090513 has an initial dose estimate of 13.590 REM and the actual dose was 29.692 REM, which results in a delta dose of 16.102 REM. However, only 10.091 REM was justified; please explain why the remaining 6.011 REM was not also justified. Is it because justifying the 10.091 REM was all that was needed to achieve less than 150% delta? This question applies to all RWPs.

For RWP 20090513, the delta of 6.011 person-rem was not explained because it corresponds to dose that was greater than the original estimate and for conditions other than what was unanticipated. The same reason is applicable to the other RWPs evaluated in section 6.2; that is, the remaining dose deltas were not explained because they correspond to dose that was greater than the original estimate and for conditions other than what was unanticipated.

4) On RWP 20090513, in the reason column, the reason associated with 6.030 REM is shown as "High RCS activity from hard reactor trip, high dose rates on insulation packages from RCP vapor seal de-staging." Is there one or two reasons being presented here? Explain the reason in more detail. (Each reason should have a verifiable dose total.)

This is meant to be one reason – the RCP vapor seal destaging. The provided reasoning was acknowledging the compounding affect from high RCS activity. That is, the magnitude of dose caused by destaging of the RCP seals is directly impacted by the magnitude of the RCS activity at the time the seals de-staged.

5) On RWP 20090600 and others it states, "Leaking RCP vapor stage seals/CRUD Plate-out in RCS piping from hard reactor trip (D-rings). Is this one or two reasons" (Each reason should have a verifiable dose total.)

Where this statement is made in RWPs 20090600, 20090601, 20090606, 20090610, and 20090618, the information is provided as one, combined reason.

The principal reason is the RCP vapor seal destaging (considered a plane source). CRUD plate-out (considered a line source) contributed to some of the dose; however, its value or contribution cannot be readily determined separately, likewise for the RCP seal destaging. Work under these RWPs was affected by both radiation sources occurring at varying intensities and totals within each of the RWPs. Because of this, independent values cannot be determined to a sufficient level of accuracy that can stand up as verifiable values. We believe the major contributor was the RCP vapor seal leakage due to its significantly higher source term and wide area impact.

A	B	C	D	E	F	G	H	I	J
RWP	**Title**	**Initial Dose Estimate (rem)**	**Actual Dose (rem)**	**Initial % Delta (D/C)**	**Justified Dose (D-G)**	**Justification - Based on RWP documentation**	**new % Delta (using justified dose) (F/C)**	**Final approved RWP Dose Estimate (rem)**	**Actual dose to Final RWP dose Estimate (%) (D/I)**
20090617	Refuel 16 Radwaste Activities	3.822	9.718	254	5.401	4.317	141	8.399	116

2009-0617, Refuel 16 Radwaste Activities

RWP	Title	Initial Dose Estimate (rem)	Actual Dose (rem)	% Delta
20090617	Refuel 16 Radwaste Activities	3.822	9.718	254

Dose estimates for this job were revised as follows:

- From 3.822 rem to 3.822 rem (revision 1)
- From 3.822 rem to 6.453 rem (revision 2)
- From 6.453 rem to 8.399 rem (revision 3)
- From 8.399 rem to 8.399 rem (revision 4)

RWP impacts (impacts realized either implementing contingency or based on unforeseen conditions)

Reason	Dose (rem)
Additional time spent decontaminating the upper reactor cavity due to high contamination levels post-cavity drain-down	1.563
Additional time spent decontaminating the -11 RCB due to leaking RCP vapor stage seals	1.657
Additional time spent cleaning the reactor vessel flange due to higher than anticipated contamination levels post cavity drain-down	.565
Additional time spent due to extension of the refuel outage	.532
Total	**4.317**