Reactor Internals

December 17, 2015

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Agenda

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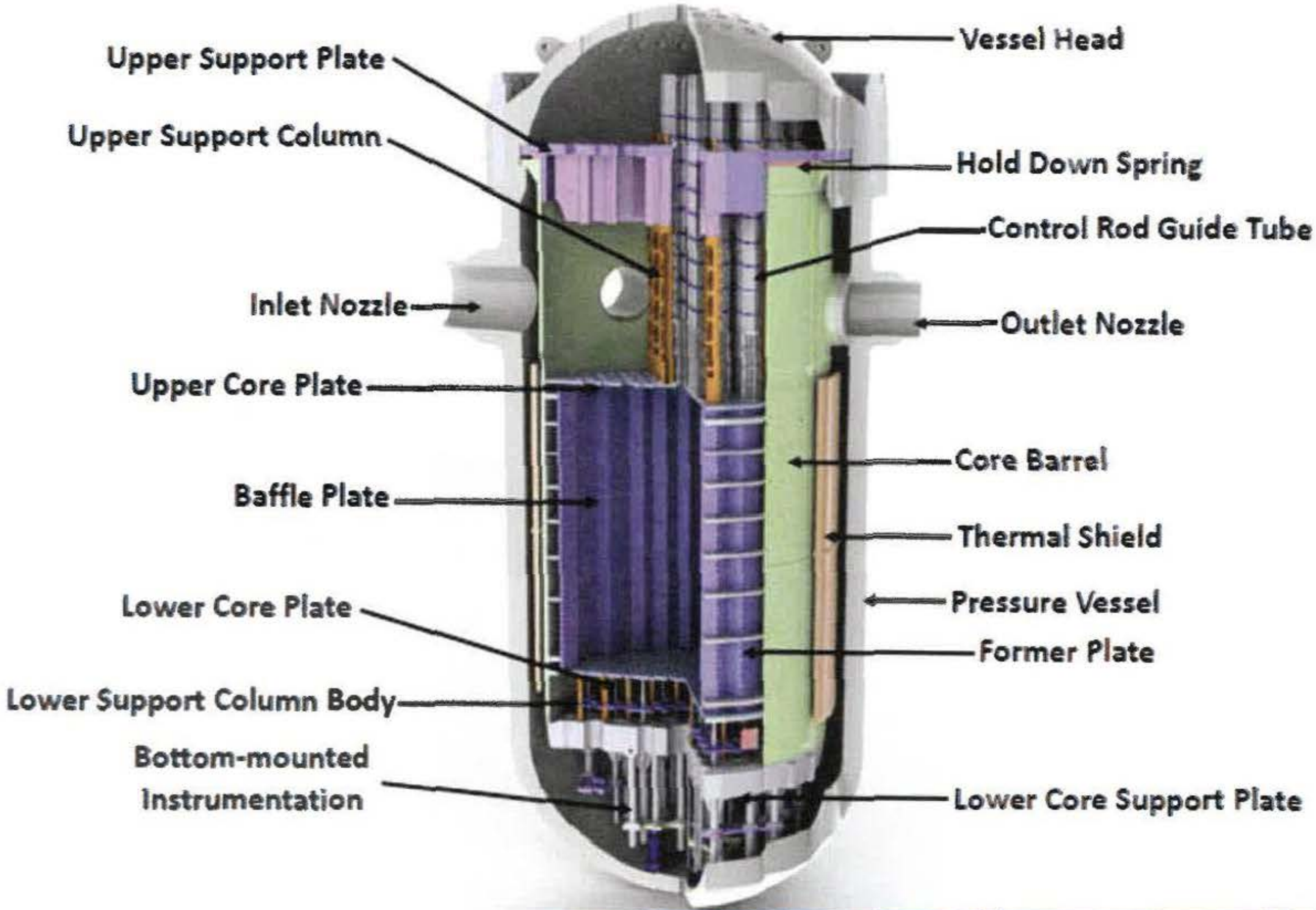
- 1. Reactor Internal Basics
- 2. Fuel Basics
- 3. Engineering Concerns for Reactor Internals
- 4. NRC Reviewer's Guidance to provide Oversight for Reactor Internals
- 5. Licensing
  - A. Initial
  - B. License Renewal
- 6. Case Study - Indian Point License Renewal for Reactor Internals

Reactor Vessel and Internals –
Typical Westinghouse-design PWR



Looking Down Into the Core Barrel During Refueling

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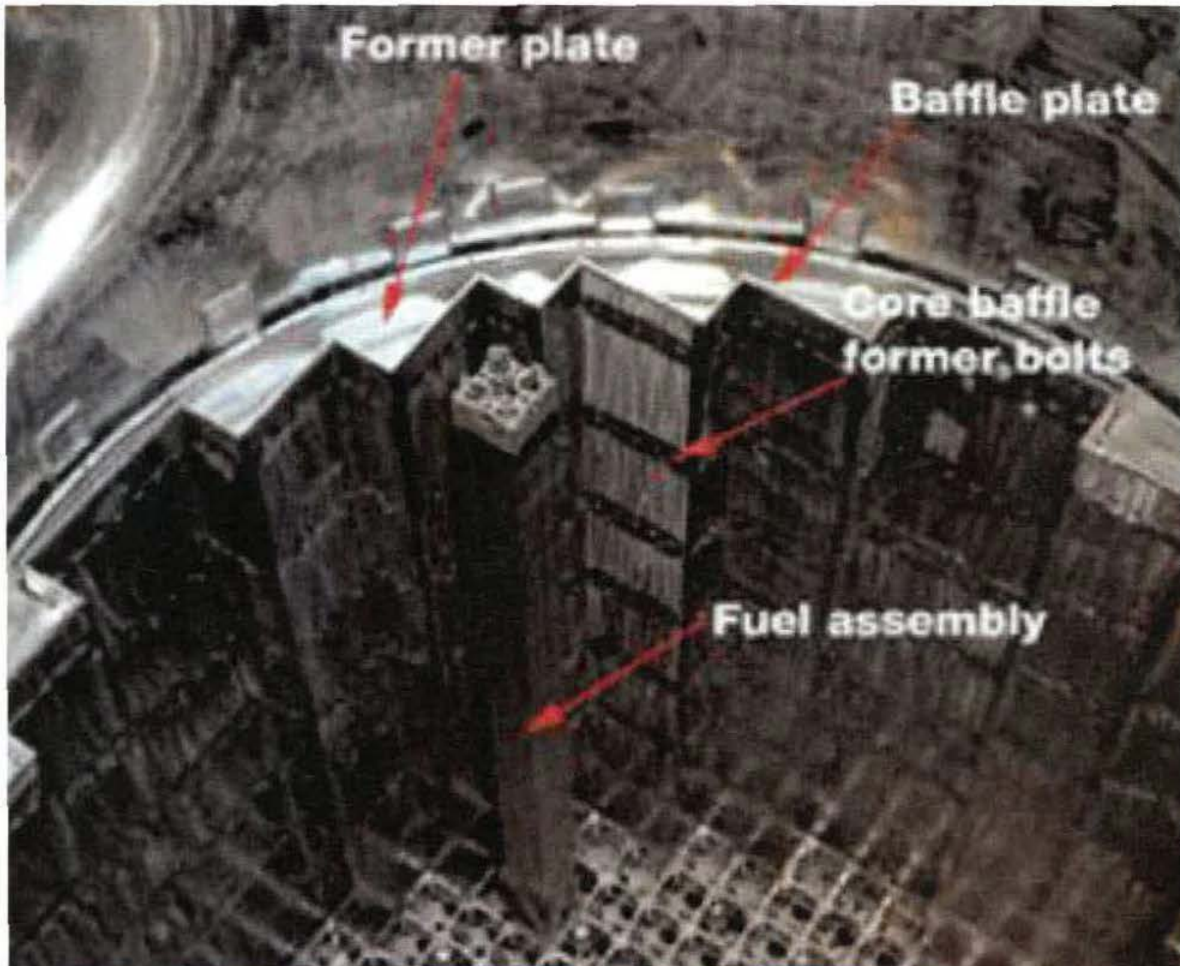

Reactor Vessel Internals Functions

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- Support core (fuel)
- Direct flow
- Guidance for control rods (safe shutdown)
- Alignment

** Key aspect of Operability and License Renewal

Pressurized Water Reactor Internals – Materials & Operating Conditions

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- Material is mainly Type 304 stainless steel
- Exposed to reactor coolant water at ~550-600 degrees F
- Reactor coolant water chemistry is carefully controlled to limit oxygen and impurities to very low levels
- High levels of neutron irradiation
- Some forms of materials degradation not seen elsewhere in the plant can occur in RVI due to the high radiation levels.

Pressurized Water Reactor Internals – Aging Effects in RVI

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- Embrittlement
  - Neutrons damage the material over time:
  - Material strength increases
  - Ductility decreases
- Stress corrosion cracking
  - Combination of a corrosive environment and sustained stress causes cracking in certain materials
  - Not usually a problem in PWR internals because of good water chemistry
- Irradiation-assisted stress corrosion cracking
  - A form of stress corrosion cracking that affects highly irradiated stainless steel

Aging Effects in RVI (cont.)

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- Fatigue
  - Repetitive stress cycles can eventually cause cracks. Usually not a problem in RVI because it is prevented by design procedures
- Void swelling
  - Materials forms voids causing a volume increase
  - Only expected in the most highly irradiated parts of internals
- Irradiation Creep
  - Under high irradiation an a constant stress, material can deform causing dimensional changes.
- Irradiation Stress Relaxation
  - Caused by the same process as irradiation creep, but causes a loss of preload in bolts instead of dimensional change.
  - Can cause bolted joints to loosen.

Pressurized Water Reactor Internals - Design of Reactor Vessel Internals

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- ASME Code, Section III did not have rules for design of RVI in early 1970's when many current operating plants were designed
- Therefore, RVI for many plants were not designed or fabricated using ASME Code rules.
- Designers tried to follow the Code to the extent possible for RVI.
- For example, Indian Point Unit 3, used the stress criteria of the ASME Code, Section III as a guide during design.

Pressurized Water Reactor - Internals ASME Code

- American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code) provides the "rulebook" for design and inspection of nuclear plant piping and vessels.
- ASME Code Sections III and XI are incorporated by reference in 10 CFR 50.55a, "Codes and Standards"
  - Section III – Rules for Construction of Nuclear Facility Component
    - Covers materials, design, fabrication, examination and testing during construction
  - Section XI – Rules for Inservice Inspection of Nuclear Power Plant Components
    - Specifies periodic inspection of piping systems and vessels
    - Also includes rules for repair and replacement, and evaluation of flaws found during inspections
- **10 CFR 50.55a requires all plants to periodically update the version of Section XI used as the basis for the ISI program**

ASME Code Cont.– Other Important Sections for Nuclear Plants

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- Section IX – Welding and Brazing Qualifications
- Section V – Nondestructive Examination
- Section II – Materials
  - **For example - Provides material properties such as allowable stress to be used in the design process**

Pressurized Water Reactor Internals - Inservice Inspection (ISI)

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- All plants have an ISI program based on ASME Code, Section XI as described in their FSAR.
- Plant systems are classified as Class 1, 2, or 3 based on safety significance.
- Reactor coolant system is generally Class 1.
- Section XI requires ultrasonic examination of welds in Class 1 piping and vessels.
  - This is done on a sample for piping welds, but typically 100% of vessel welds.
- Examinations are done on a ten-year interval.
- For reactor internals, only visual examination is required by Section XI.
- Visual examination for general structural and mechanical condition, called VT-3 by the Code.

Pressurized Water Reactor Internals - Licensing Basics

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- Commercial power reactors are licensed under
  - 10 CFR 50 (currently operating)
  - 10 CFR 52 (new reactors – combined operating license)
- Original operating license for all reactors is for 40 years.
- **License renewal – Licensees may apply to renew the license for 20 more years under 10 CFR 54.**

Pressurized Water Reactor Internals - License Renewal

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- Not all systems, structures and components (SSCs) are within the scope of 10 CFR 54.
- In scope SSCs are:
  - safety related
    - safe shutdown
    - reactor coolant pressure boundary
    - Prevent offsite radiation exposures in excess of certain limits
  - Non-safety related whose failure could impact safety function of safety related components
  - Needed for other regulated processes, such as fire protection or station blackout

Pressurized Water Reactor Internals - License Renewal

- One of the key requirements for the Commission to issue a renewed license is that the applicant must provide reasonable assurance that the aging effects are managed, such that there is reasonable assurance that the systems, structures and components can perform their "intended functions" until the end of the 20-year license extension.
- "Aging Effects" include corrosion, cracking, fatigue, and embrittlement
- Embrittlement of a material means it takes less energy to break
- Exposure of steel to neutron irradiation from the reactor causes the steel to become embrittled over time.
- "Effects" vs "Mechanism"

LR In-Scope Components Subject to Aging Management

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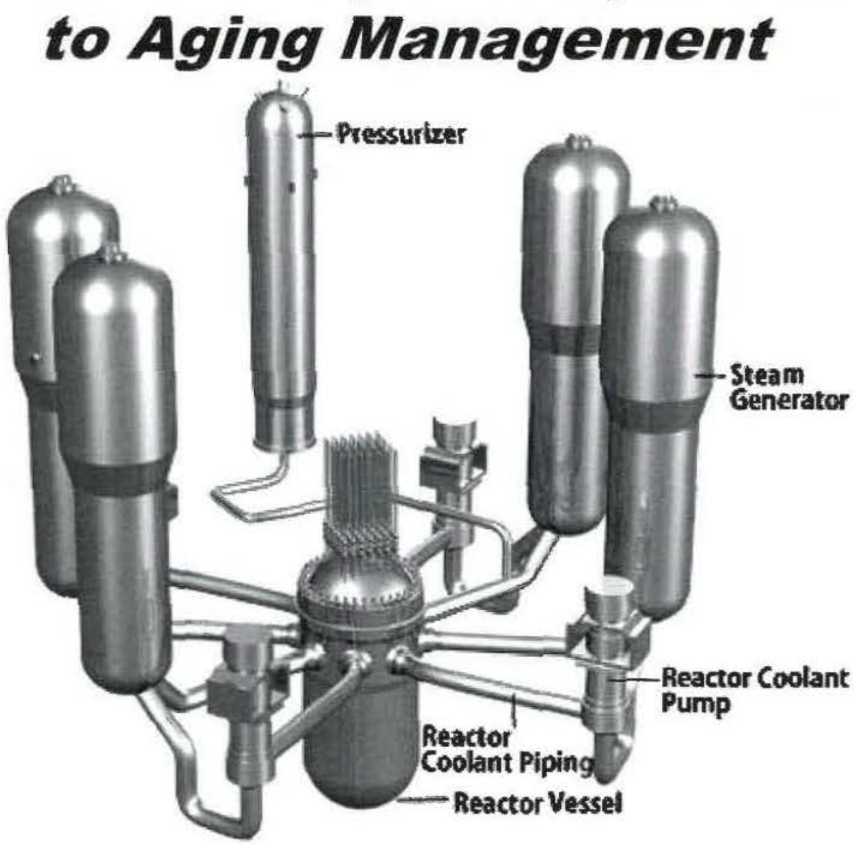

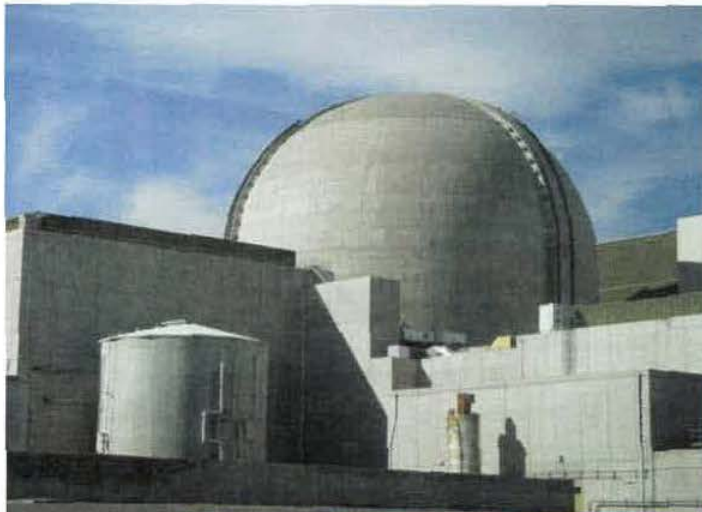

License Renewal – What components are subject to aging management?

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- Only systems, structures or components (SSC) that are "passive" and "long-lived" are subject to aging management.
  - "Passive" means an SSC performs its function without moving parts or a change in configuration or properties
    - Example –piping, reactor vessel, containment building, valve and pump bodies
    - However, valve internals such as stems and discs are excluded, because they move, as are pump impellers.
    - In reactor, control rods are not subject to aging management because they are neither passive nor long lived.
  - "Long Lived" components are not subject to replacement based on a qualified life or specified time period. In other words, things expected to last the life of the plant.

Current Licensing Basis

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- *Per* § 54.3 *Current licensing basis* (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 52, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant-specific design-basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

Time-Limited Aging Analyses (TLAA) per § 54.3:

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- (1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);
- (2) Consider the effects of aging;
- (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
- (4) Were determined to be relevant by the licensee in making a safety determination;
- (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and
- (6) Are contained or incorporated by reference in the Current Licensing Basis (CLB.)

License Renewal and Current Licensing Basis

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- For license renewal, applicants need to show the CLB will be maintained for the period of extended operation, but are not required to make changes to the CLB.
- For example, a structural analysis of the RVI is described in the IP2 UFSAR.
- This analysis does not consider the properties of the RVI after irradiation, and has no time-limited assumptions.
- This analysis is therefore not TLAA.
- The applicant does not have to revise the RVI structural analysis to consider irradiated properties of the steel for license renewal.
- However, if the original RVI structural analysis had considered the effects of irradiation for 40 years, that analysis would have to be revised for 60 years, shown to be good for 60 years, or the aging due to irradiation could be managed by an aging management program.

Aging Management Programs

- Applicants may credit an aging management program to manage aging effects
- Programs may be existing plant programs such as ISI or new programs just for license renewal
- Programs may rely on inspection such as the ISI program, or may be preventive, such as Water Chemistry.
- For RVI, staff determined ISI program wasn't sufficient by itself to manage all the aging effects.

License Renewal Guidance

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- NUREG-1801, Generic Aging Lessons Learned Report (GALL), provides guidance for
  - aging management review – what aging effects should be managed for specific SSCs, and what programs manage them?
  - Analysis of Time-Limited Aging Analyses
  - Aging Management Programs
- NUREG-1800, Standard Review Plan for License Renewal
- Staff periodically revises these documents.
- Staff may issue Interim Staff Guidance (ISG) if portions of GALL or SRP-LR need to be revised between major revisions of the NUREGs.
- NRC guidance is considered one acceptable method for applicants to comply with 10 CFR 54; however, it is not regulation and applicants may use other methods.

Industry RVI Aging Management Program

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- The industry, led by the Electric Power Research Institute (EPRI), began development of an aging management program for PWR internals around year 2000.
- Industry program needed results from materials testing that was not complete yet
- NRC staff allowed applicants for renewed licenses to make a commitment to implement the industry program once it was issued.
- The report describing industry's recommended program was submitted to NRC for review and approval on 1/12/2009.
  - **MRP-227, Rev. 0, Materials Reliability Program: Pressurized Water Reactor Internals Inspections and Evaluation Guidelines**

MRP-227

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- MRP-227, Rev. 0 is primarily an inspection-based program.
- Prescribes inspections of certain components of the RVI using different methods than specified by the ASME Code
  - Higher-resolution visual examination (EVT-1)
  - Ultrasonic examination (UT)
- Components inspected based on likelihood and safety impact of aging
- Specifies inspections for three different generic designs: Babcock & Wilcox (B&W), Combustion Engineering, and Westinghouse.
- NRC staff approved MRP-227 with several conditions and action items, in safety evaluation dated 12/16/11.
  - Approved for applicants/licensee to use as the basis of plant-specific RVI AMPs
  - Applicants or licensees must completed the applicable action items

Examples of RVI Component Inspections – Baffle-Former Bolts (UT)

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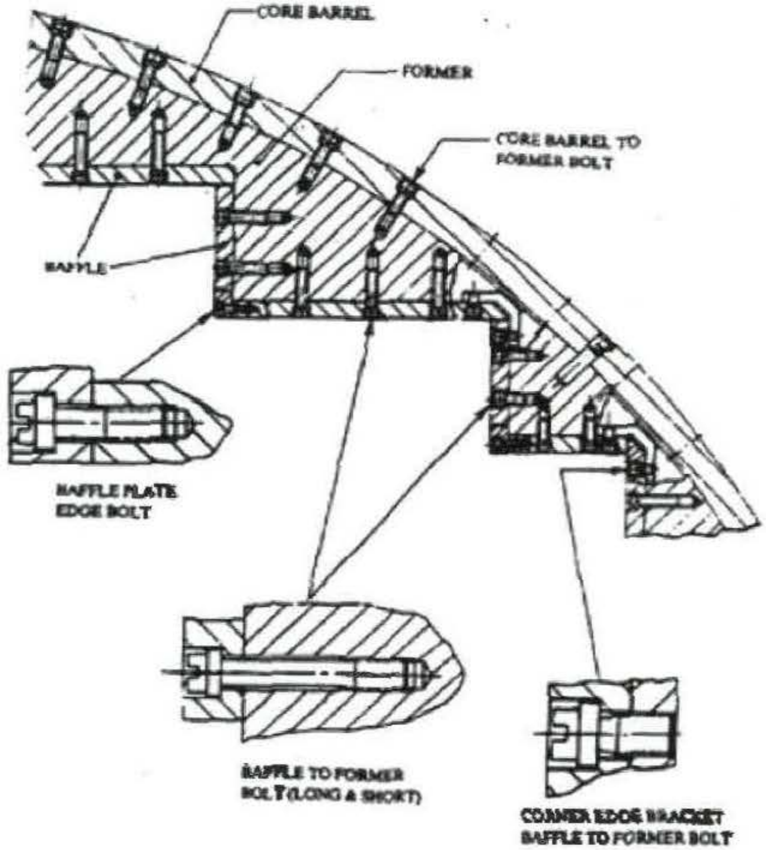
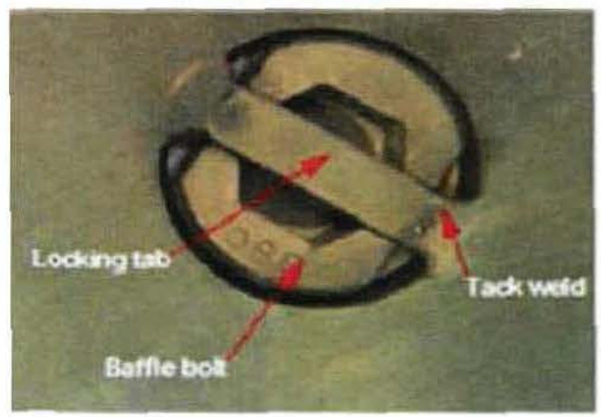
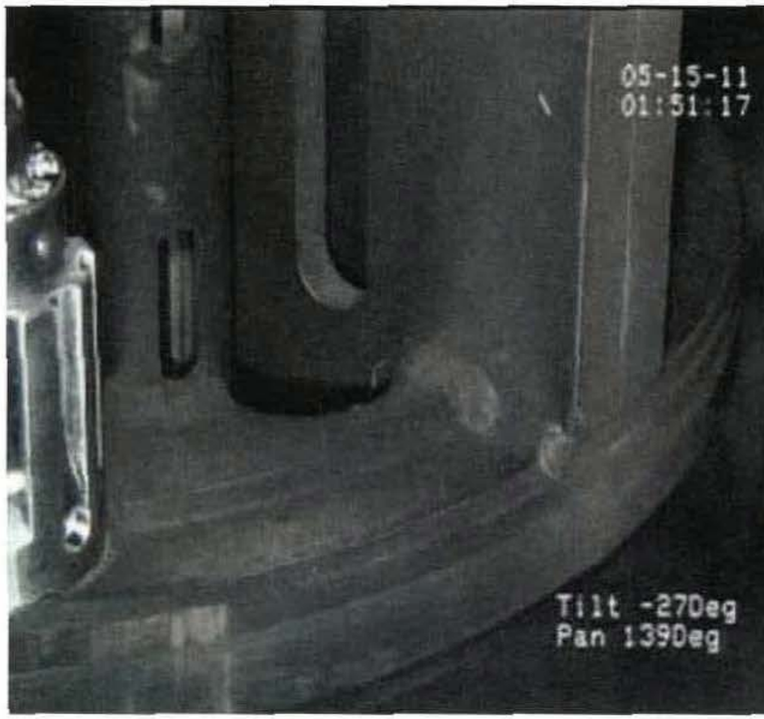


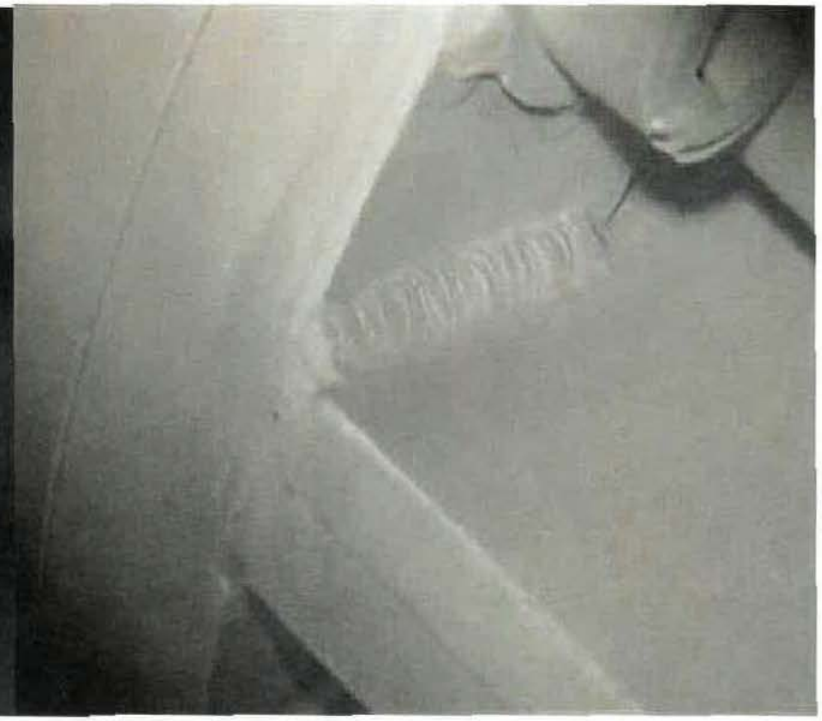
External Hex

Internal Hex

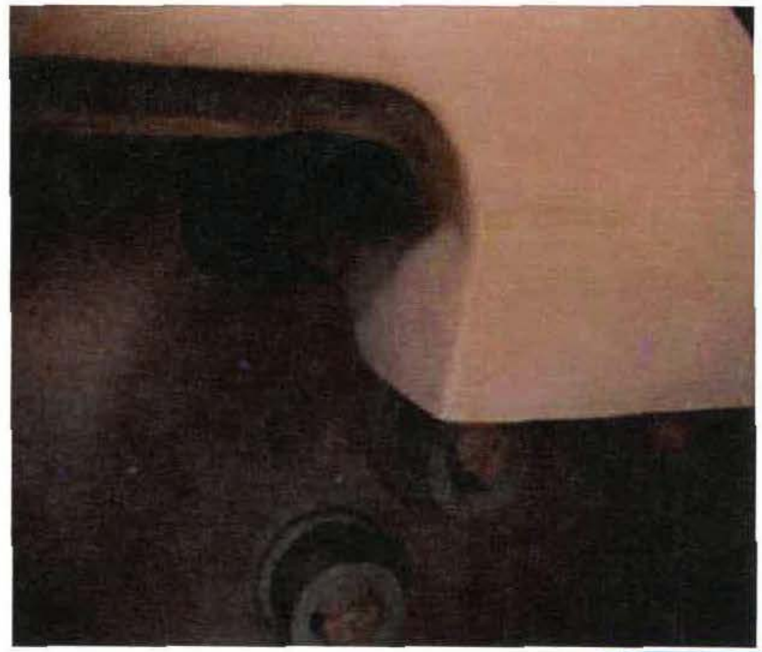


Examples of RVI Component Inspections – Control Rod Guide Tube Lower Flange (EVT-1)





Thermal Shield Flexures (VT-3)



Safety Evaluation Report – License Renewal

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- NRC develops a safety evaluation report (SER) to document its review of a license renewal application.
- Staff reviews the LRA against criteria in the GALL report and SRP-LR, and the regulations (10 CFR 54).
- NRC staff may issue Requests for Additional Information (RAI) to the applicant when it needs more information to make a safety determination.
- NRC staff may perform audits, particularly to verify applicant's claims that AMPs are consistent with GALL.

§ *2.309(f)(1) Says to be admitted a contention must:*

- (i) Provide a specific statement of the issue of law or fact to be raised or controverted . . .
- (ii) Provide a brief explanation of the basis for the contention;
- (iii) Demonstrate that the issue raised in the contention is within the scope of the proceeding;
- (iv) Demonstrate that the issue raised in the contention is material to the findings the NRC must make to support the action that is involved in the proceeding;
- (v) Provide a concise statement of the alleged facts or expert opinions which support the requestor's/petitioner's position on the issue . . . ; [and]
- (vi) . . . . [P]rovide sufficient information to show that a genuine dispute exists with the applicant/licensee on a material issue of law or fact.
  - **I.e., you can't just say "Nuclear is bad"**

§ 2.309 – Requirement for Standing

- Parties filing contentions must have "standing"
- Key factors for the ASLB to consider with respect to standing are:
  - The nature and extent of the requestor's/petitioner's property, financial or other interests in the proceeding; and
  - The possible effect of any decision or order that may be issued in the proceeding on the requestor's/petitioner's interest;

Conduct of License Renewal Hearings

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- Three-judge panel
- Prefiled written testimony is filed by all parties.
- Hearing is conducted similar to a trial
- Expert witnesses testify for each party
- Attorneys are present for each party
- Judges ask questions of the witnesses
- Judges may or may not allow cross examination
- The ASLB will rule on the contentions after the hearing.
- The Commission reviews and may reverse the ASLB ruling.
- Parties may appeal.

Case Study – Indian Point

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- SER for LR of IP2 and IP3, NUREG-1930, was issued in August 2009 after a 28-month review period.
- Normally, the commission would issue the renewed license within about 6 months of the SER publication.
- Due to contentions, IP2 and IP3 renewed license has been delayed.
- For IP2 and IP3, after the LR SER was published, but before a renewed license was issued, the applicant submitted new information.
- The new information was an amendment to the LRA containing a RVI AMP based on MRP-227, Rev. 0.

Case Study – Indian Point

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- Since the staff had not yet approved MRP-227, Rev. 0, the review of the amendment went through several iterations.
- Applicant also submitted its RVI Inspection Plan on 9/28/2011, 24 months before IP2 end of original 40-year license, to fulfill its commitment.
- A revised RVI inspection plan submitted on 2/17/2012 consistent with NRC-approved version of MRP-227, published 1/9/2012. (MRP-227-A)
- NRC reviewed RVI AMP and Inspection Plan using Interim Staff Guidance that updated Rev. 2 of GALL to reference MRP-227-A.
- NRC issued Supplement 2 to the SER on 11/7/2014 to document its review and approval of IP2 and IP3 RVI AMP and Inspection Plan.

Hearings

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Case Study – Indian Point Hearings

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- Atomic Energy Act Section 189 provides allows Commission to grant a hearing for granting, suspending, revoking or amending any license
- Atomic Energy Act Section 191 establishes the Atomic Safety and Licensing Board and authorizes it to conduct such hearings
- § 54.27 allows for hearings under §2.309, "Hearing requests, petitions to intervene, requirements for standing, and contentions"
  - Under § 2.309, Any person whose interest may be affected by a proceeding and who desires to participate as a party must file a written request for hearing and a specification of the contentions which the person seeks to have litigated in the hearing.
- ASLB also conducts the hearings related to granting licenses and amendments under § 50, petitions under §2.206. All hearings are conducted under the rules of §2.

Case Study – Indian Point Contention NYS-25

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- Original: Entergy's license renewal application does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor pressure vessels ("RPVs") and the associated internals.
  - The LRA does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the reactor pressure vessels. ("RPVs") and the associated internals at both plants, pursuant to 10 C.F.R. § 54.21 (a), and an evaluation of time limited aging analysis, pursuant to 10 C.F.R. § 54.21(c).
- Amended – Acknowledged that Entergy has a program for managing aging of reactor vessel internals, but argues that it is inadequate. Focus shifted away from RPV to RVI.
- 2nd Amendment – After staff SER was published, new information available, so ASLB admitted

Case Study – Indian Point Contention NYS-25

- One of New York's bases for NYS-25 was that Entergy should revise the RVI structural analyses to consider embrittlement.
- This is not required by § 54 since the analysis in the CLB didn't consider embrittlement.
- Other bases included:
  - Failure to considered combined or "synergistic" effects of multiple aging effects, particularly fatigue and embrittlement
  - Inadequacy of inspection methods,
  - Components not being preventively replaced
  - Fatigue calculations did not include an error analysis

Case Study – Indian Point Hearing Aftermath and Timely Renewal

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- Hearing on NYS-25, plus two other contentions, held November 16-20, 2015 in Tarrytown NY.
- The ASLB will probably take several months to issue its decision on NYS-25 and the other contentions.
- Due to contentions, the commission has still not made a decision on whether to issue a renewed license for IP2 and IP3
- Meanwhile, the original license for IP2 expired on 9/28/2013, and for IP3 expired on 12/12/2015.
- Because the NRC has not made a final determination on whether to issue a renewed license, under the regulation called "timely renewal", the plants are allowed to continue to operate until the NRC makes a final decision.

Case Study – Indian Point IP2&IP3 License Renewal Timeline (Backup)

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- License renewal application (LRA) submitted – 4/30/2007
  - Contained commitment to implement industry program for reactor vessel internals
- New York filed petition to intervene – 11/30/2007
- ASLB Admitted
- Industry guidance published for RVI inspection – MRP-227, Rev. 0 – 1/12/2009
- Safety Evaluation Report (SER) issued – 8/11/2009
- Amendment 9 to LRA submitted to NRC – 7/14/2010
  - Contained aging management program for reactor vessel internals consistent with MRP-227, Rev. 0
- New York filed motion to add more bases to NYS-25 – 9/15/2010

IP2&IP3 License Renewal Timeline

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- ASLB granted New York's motion to add bases NYS-25 – 7/6/2011
- NRC Staff Final Safety Evaluation of MRP-227, Rev. 0 – 12/16/11
- Entergy submits RVI inspection plan consistent with NRC Final SE – 2/17/2012
- NRC staff issues Supplement 2 to Safety Evaluation Report for License Renewal, in which the staff approved the RVI program – 11/7/2014
  - Published as NUREG-1930, Supplement 2 – 7/31/2015
- New York filed motion to add more based to NYS-25 – 2/13/2015
- ASLB granted motion to add bases – 3/31/2015
- Hearing on Contention NYS-25 could not be scheduled until staff made its finding on the RVI program for IP2 and IP3.
- Hearing was held November, 16-21, 2015, in Tarrytown, NY (along with two other contentions).

OpE COMM

Operating Experience Communication

Good Judgment Comes from Experience

Information Security Reminder - OpE COMMs contain preliminary information in the interest of timely internal communication of operating experience. This information is often subject to change and is not intended for distribution outside the agency in this form.

St. Lucie Unit 1 – Rebuilt MSIV Fails After 230 Days in Use

Executive Summary:
To support an Extended Power Uprate (EPU), new internals were installed in both Main Steam Isolation Valves (MSIV). It was not known at the time, but new tail links did not meet design specification dimensional drawings (they were oversized). The oversized tail links did not allow the valves to backseat fully, leaving the disc partially in the flow stream. This exposed the valves to unintentional dynamic loading, which ultimately resulted in the failure of internal parts. For the "B" MSIV, the failure caused the valve disc to seat uncontrollably while at 100% power, tripping the reactor. The direct cause was that the valve disc was not backseated which allowed unintentional loading of internal parts. The root causes were that the tail links did not meet design specification dimensional drawings and that the engineering change package did not include a verification that the modified valves would open fully.

COMM Groups Notified:
All Communications, Inspection Programs, Main Steam & Condensate/Feed Systems, Materials/Aging, New Reactors, Power Uprate, Pump and Valve Performance, and Vendor & Quality Assurance.

Background:

St. Lucie Units 1 and 2 are two loop Combustion Engineering plants. St. Lucie 1 received a Power Uprate approval on July 9, 2012 (ML12191A220). The approval authorized a 10.0% extended power uprate (EPU) and a 1.7% measurement uncertainty recapture, increasing the maximum steady-state reactor core power level from 2700 megawatts thermal (MWt) to 3020 MWt.

To support the approved higher power levels, the Main Steam Isolation Valves (MSIV) internals had to be upgraded to insure they could withstand the impact stresses associated with spurious closure at the higher steam velocities. It was determined that the original MSIV body (Schutte and Koerting, Co (now Ametek, Inc.), model M70-00656-V) could be retained and did not need to be replaced. Kalsi engineering provided the design and analysis (Engineering Change Package EC246556) with an independent third party design review by Zachary Engineering. Zachary Engineering (a separate group from the independent design analysis) provided the installation and post maintenance testing requirements. Flowserve manufactured and performed the field work to install the new internal valve components. Enertech provided and installed a new electro-hydraulic actuator to replace the existing pneumatic actuator. Bechtel provided QA services at Flowserve during parts manufacture. [NOTE: the licensee's root cause evaluation noted that this complex project structure created an error likely environment.]

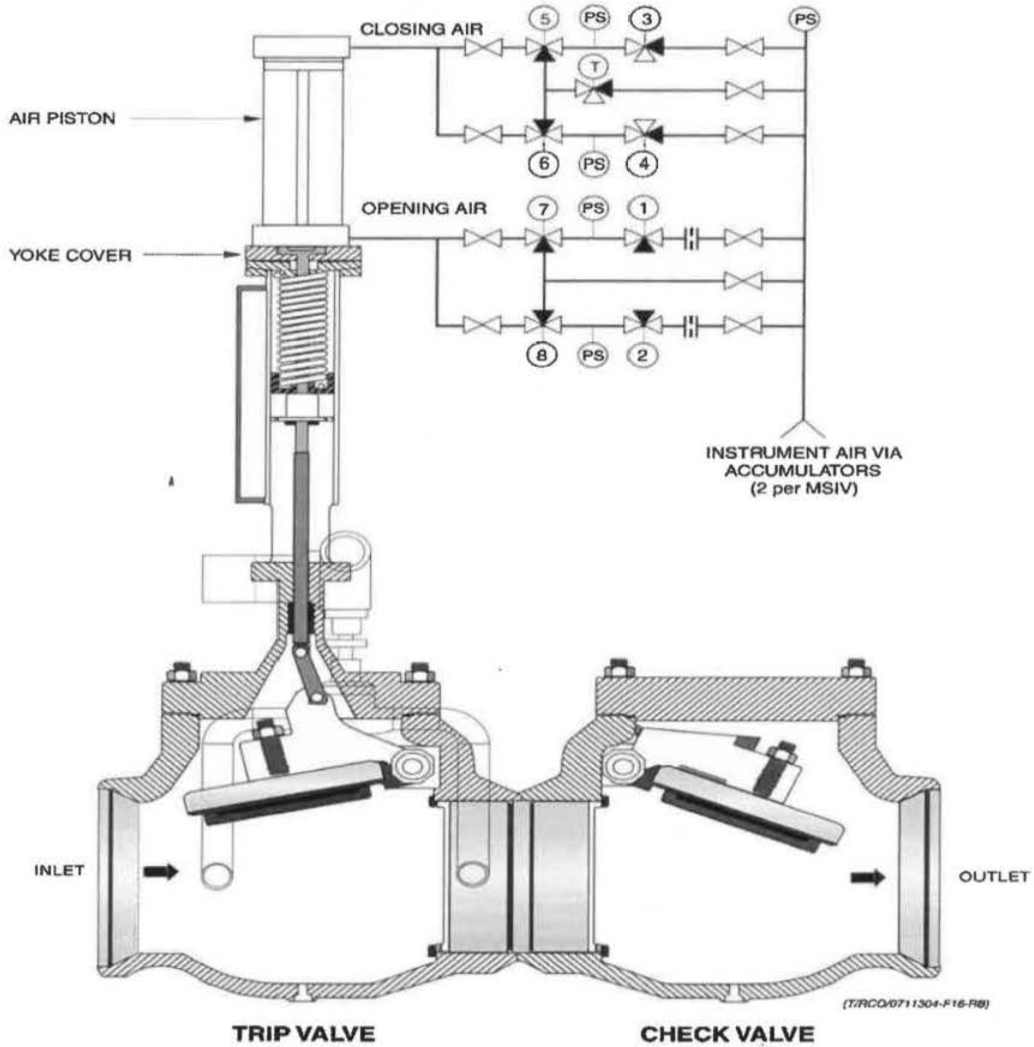


Figure 1
St. Lucie Unit 1 Main Steam Isolation Valve
Shown in the Open Position

The new internals and actuator were installed during a planned shutdown on July 18, 2012 (See Figure 1). The work orders that installed the internals and actuator required stroke length measurements and provided acceptance criteria (as specified in the modification acceptance test plan). This was identified as a critical step to insure the valve would stroke fully closed to fully open. The work orders were closed even though the acceptance criteria were not satisfied for either MSIV. FPL procedures require an action request (AR) for "unexpected or unwanted conditions". The discrepant stroke length was documented, but no AR was initiated.

During post installation limit switch adjustments, the limit switches could not be set within existing limits. Change request notices (CRN) documented this. The engineering resolution to the CRNs focused on limit switch set point adjustments and limit switch mounting changes, and did not address why the changes were needed. (Later investigation determined that this was a missed opportunity to detect the problem. A more thorough engineering review may have prevented the problem.)

From July 26 through July 29, startup testing was performed, and unit 1 reached the new 100% power level on July 29, 2012.

Issue Description:

On March 12, 2013, after 230 days on line at the new rated power level, St. Lucie Unit 1 tripped on thermal margin/low pressure (TMLP) from an asymmetric steam generator (SG) transient (275 psid between the SGs) signal (See EN 48818). After the trip, the licensee determined that, with the valve positioned to "OPEN", the valve did not pass enough flow to maintain steam header pressure. Steam header pressure could only be maintained with the MSIV bypass valve in service. The licensee cooled down to Mode 5 to inspect the MSIVs internally. Internal inspection found that the "B" MSIV had a spindle to disc separation. A similar inspection of the "A" MSIV found the disc still attached to the spindle, but the spindle itself was damaged. See Figure 2 below.

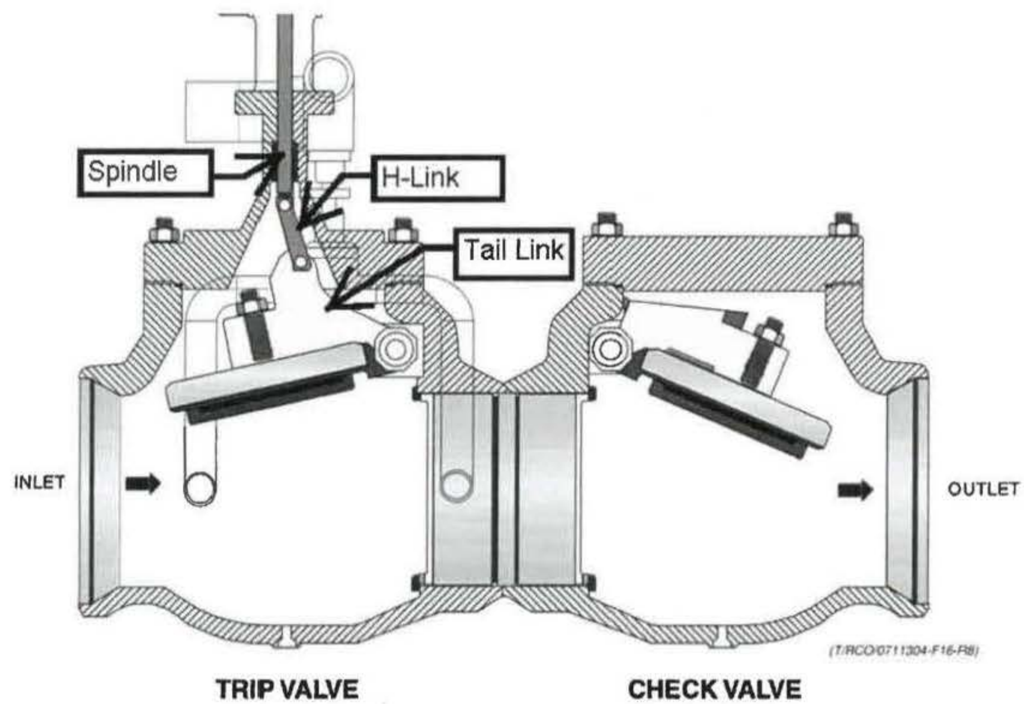


Figure 2
MSIV Close Up Showing Tail Link, H-Link and Spindle

The license discovered that the back of the tail link for both MSIVs made contact with the valve body before the back seat stopped valve motion (notice the worn portion of the tail link in Figure 3). The tail link was dimensionally checked against the design requirements, and determined to be oversized.

Figure 3
MSIV Tail Link

With the disc not fully backseated, it remained partially exposed to the steam flow through the MSIV. Excessive stress was caused by the valve disc remaining partially in the flow stream. These stresses caused the shear pin for the "B" MSIV to fail (See Figure 4). When the shear pin failed, the spindle separated from the disc. System flow then uncontrollably forced the valve disc into the seat, blocking flow from the steam generator (Inlet side) to the main steam system (Outlet side).

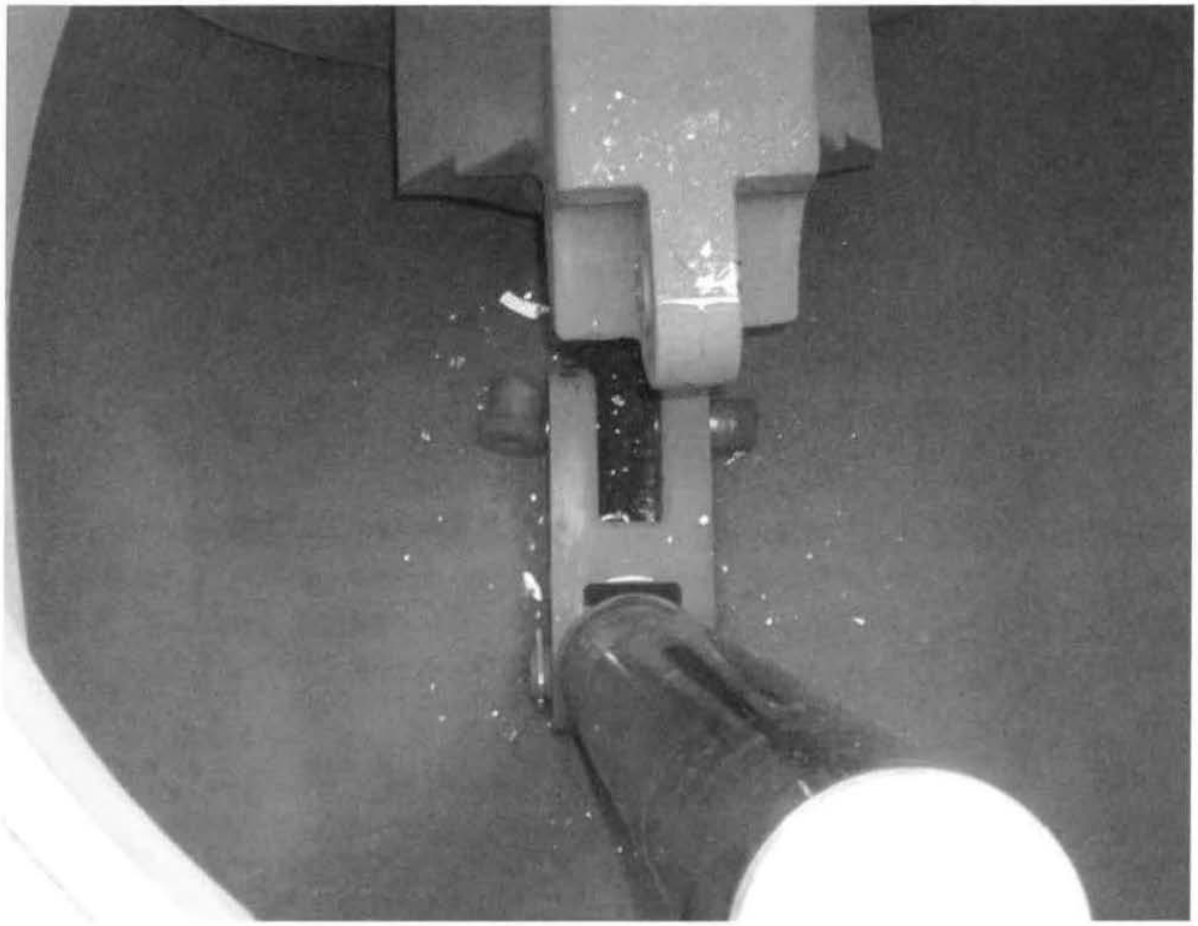


Figure 4
"B" MSIV Broken Shear Pin

They also discovered the same oversized tail link and interference on the "A" MSIV. However, instead of the shear pin failure seen on the "B" MSIV, the spindle tab had partially torn (See Figure 5). The licensee determined that, once the partial tear occurred on the spindle tab, the "A" MSIV disc fully backseated, which prevented the in-service failure.

Figure 5
"A" MSIV Damaged Spindle Connection

When the back of the tail link made contact with the valve body prior to the valve being fully backseated, the valve disc was not fully in its open position. When fully opened and backseated, the valve disc must be open at least 80 degrees with a 1/8 inch clearance to the body to account for thermal expansion. The newly installed tail link was oversized, making contact with the valve body before the seat was backseated. This caused excessive, unintended stress on the valve components, ultimately resulting in the shear pin failure on the "B" MSIV causing the disc to separate from the spindle. When the pin failed on the "B" MSIV, the valve shut uncontrollably.

The excessive stresses imposed on the valve from the uncontrolled shutting of the MSIV from 100% power required extensive engineering analysis and replacement of several internal components, including the valve seat, and some external supports. The engineering analysis determined that the valve body was acceptable for continued use.

The licensee determined that there were 2 root causes for this event:

1. The tail links provided by Flowserve did not meet the design specification dimensional requirements. The contract with Flowserve required all parts be provided per detailed machine drawings. Shop inspections and dimensioning did not identify the tail link discrepancies. QA inspections by Bechtel did not

identify the dimensional discrepancies. FPL QA surveillance was waived for this hold point, as allowed by FPL procedures. This cause was characterized as a personnel performance issue.

2. The engineering change package did not include verification that the modified valve would open fully, i.e., to the back seat. The EC process includes requirements to specify required implementation instructions and post maintenance testing. Although multiple engineering groups reviewed the EC, the package did not contain specific instructions to assure the valve is fully open to the backseat. This cause was characterized as a personnel performance issue.

Licensee Event Report 3352013001R0 (ML13142A200) was issued by the licensee on March 12, 2013.

The licensee completed an extent of condition review and determined that no other valves were affected. It should be noted that the Unit 2 MSIVs are a completely different design. The Unit 2 MSIVs are 34 inch Rockwell International (Model Number PD-153115) angled valves with an internal pilot and balance chamber. Licensee discussions with Flowserve indicate that they don't believe that any other plants have had this type of modification performed.

If anyone has additional information related to this OpE COMM, please contact Bob Bernardo at Robert.Bernardo@nrc.gov or 301-415-2621.

The following insights associated with this event apply to operating reactor and new construction inspection activities:

1. Oversight: Adequate oversight and inspection are critical to ensuring quality. In this case, the licensee established a complex project organization to complete this design change that relied on the use of multiple vendors. The inspections performed did not ensure that the tail links provided by Flowserve met the design specification dimensional requirements, as required by Criterion X of 10 CFR 50 Appendix B. Appendix 10 of IP 35007, "Quality Assurance Program Implementation During Construction and Pre-Construction Activities", IP 71111.18, "Plant Modifications" and section 03.10 of IP 43002, "Routine Inspections of Nuclear Vendors", provide guidance for inspecting independent oversight.

2. Testing: Per Criterion XI of 10 CFR 50 Appendix B, verification testing must incorporate the requirements and acceptance limits contained in applicable design documents. Testing anomalies, whether at the vendor shop or in-situ, must be thoroughly investigated and dispositioned. In this case, post modification testing to verify that the MSIVs would fully open was not specified in the EC Package. In addition, the anomaly caused by the stroke length measurements not meeting the acceptance criterion defined by the work order instructions was not thoroughly investigated. IP-65001.D, "Inspection of the ITAAC-Related Operational Testing Program", IMC 2513, "Light Water Reactor Inspection Program - Preoperational Testing And Operational Preparedness Phase", IP 71111.19, "Post Maintenance Testing" and IP 43002, "Routine Inspections of Nuclear Vendors", provide guidance for inspecting testing associated with construction activities and post maintenance testing for operating reactors.

Other Operating Experience:

Public Web Page on Power Uprates: http://www.nrc.gov/reactors/operating/licensing/power-uprates.html

Previous OpE COMMS:

Main Steam Isolation Valve Failures at Farley Unit 1
http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=889

MSIV inoperable due to internal binding at McGuire:
http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=1049

Main Steam Isolation Valves Fail to Close (SIT) at Harris:
http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=3841

Two MSIV Steam Failures Due to Thermal Embrittlement at Vogtle:
http://nrr10.nrc.gov/forum/forumtopic.cfm?selectedForum=03&forumId=AllComm&topicId=4012

Issue for Resolution (IFR) 2005-069: Davis Besse - Potential Inoperability of AOVs to Function During Design Basis Conditions: http://nrr10.nrc.gov/ope-info-gateway/ifr/2005/IFR%202005-069.pdf

Previous Generic Communications

NRC Information Notice 2004-15: Dual-Unit Scram at Peach Bottom Units 2 and 3 (MSIV failed to Close on Demand)
NRC Information Notice 1995-04: Excessive Cooldown and Depressurization of the Reactor Coolant System Following a Loss Of Offsite Power (MSIVs for steam generators A and B failed to close fully)
NRC Information Notice 1994-44: Main Steam Line Isolation Valve Failure to Close on Demand Because of Inadequate Maintenance and Testing
NRC Information Notice 1994-08: Potential for Surveillance Testing to Fail to Detect an Inoperable Main Steam Isolation Valve
NRC Information Notice 1990-79: Failures of Main Steam Isolation Check Valves Resulting in Disc Separation
NRC Information Notice 1988-59: Main Steam Isolation Valve Guide Rail Failure at Waterford Unit 3
NRC Information Notice 1988-51: Failures of Main Steam Isolation Valves
NRC Information Notice 1986-106: Feedwater Line Break (The event was initiated by the main steam isolation valve on steam generator C failing closed) – 4 fatalities
NRC Information Notice 1986-81: Broken Inner-External Closure Springs on Atwood & Morrill Main Steam Isolation Valves
NRC Information Notice 1986-57: Operating Problems With Solenoid Operated Valves at Nuclear Power Plants
NRC Information Notice 1985-84: Inadequate Inservice Testing of Main Steam Isolation Valves
NRC Information Notice 1985-59: Valve Stem Corrosion Failures
NRC Information Notice 1985-21: Main Steam Isolation Valve Closure Logic
NRC Information Notice 1984-35: BWR Post-Scram Drywell Pressurization (inboard main steam isolation valve (MSIV) failed closed)
NRC Information Notice 1984-33: Main Steam Safety Valve Failures Caused by Failed Cotter Pins
NRC Information Notice 1983-70, Supplement 1: Vibration-Induced Valve Failures
NRC Information Notice 1983-54: Common Mode Failure of Main Steam Isolation Non Return Check Valves
NRC Information Notice 1982-23: BWR Main Steam Isolation Valve Leakage

NRC Information Notice 1981-38: Potentially Significant Equipment Failures Resulting from Contamination of Air-Operated Systems
NRC Information Notice 1981-14: Potential Overstress of Shafts on Fisher Series 9200 Butterfly Valves with Expandable T Rings (maximum shaft stress underestimated)
IE Circular 1981-14: Main Steam Isolation Valve Failures to Close
NRC Information Notice 1980-16: Shaft Seal Packing Causes Binding in Main Steam Swing Disc Check and Isolation Valves

(b)(4), (b)(7)(D)

[redacted]

Licensee Event Reports (LER):

LER Link	Date	Site	Description
4242012005	10/08/2012	Vogtle 1	Main Steam Isolation Valve Failure
4002012001	04/21/2012	Harris	Delayed Closure of Main Steam Isolation Valves due to Corrosion
2372011003	10/17/2011	Dresden 2	MSIV Closure Times outside of Technical Specifications Limits
2492010002	11/01/2010	Dresden 3	MSIV Leakage Exceeds Technical Specifications Allowable Limits
3482006002	04/08/2006	Farley 1	Main Steam Isolation Valve Failure to Close
3702005005	03/02/2005	McGuire 2	Failure of Main Steam Line Isolation Valve (MSIV) to Close
2192004005	09/11/2004	Oyster Creek	Main Steam Isolation Valve Failed to Close During Partial Valve Closure Surveillance Due to Mechanical Binding
4131996008	03/07/1995	Catawba 1	Closure Response Time Exceeded For Main Steam Isolation Valve 1SM1, B Train
2661992006	05/31/1992	Point Beach 1	Failure of Main Steam Isolation Valve 1MS-2018 to Fully Shut During Performance of IT-280
3011991001	09/29/1991	Point Beach 2	Failure of Main Steam Isolation Valves to Close
4581989043	12/01/1989	River Bend	Failure of Two Outboard Main Steam Isolation Valves
2371988012	05/17/1988	Dresden 2	Main Steam Isolation Valves Failure to Close Due to High Stem Drag Forces Caused by Valve Packing

LER Link	Date	Site	Description
2131986029	06/17/1986	Haddam Neck	Main Steam Isolation Valve Closure Test Failure
2451986006	02/06/1986	Millstone 1	Failure of 1-MS-1D and 1-MS 2D to Close
2501986005	01/27/1986	Turkey Point 3	Main Steam Isolation Valve
3331985027	11/22/1985	Fitzpatrick	Main Steam Isolation Valve Failures
3241985008	09/27/1985	Brunswick 2	Failure of Main Steam Line Isolation Valves B21-F028A, F022C, and F028C to Fast Close During Operability Testing
3181985008	07/24/1985	Calvert Cliffs 2	Failure of MSIV to Fully Close During Surveillance Testing
4541985027	03/14/1985	Byron 1	Failure of MSIV to Close on MS Isolation Signal
3171984019	12/12/1984	Calvert Cliffs 1	Failure of #12 MSIV to Fully Close During Surveillance Testing
1551984013	09/09/1984	Big Rock Point	Failure of Main Steam Isolation Valve (MO-7050) to Close
3181982050	10/17/1982	Calvert Cliffs 2	Update on MSIV Failure to Close
2711981027	10/16/1981	Vermont Yankee	MSIV Failed to Close
2711981022	08/01/1981	Vermont Yankee	The MSIV Failed to Close
3661981071	07/18/1981	Hatch 2	Failure of the "D" MSIV to Close within Required Time Frame

Ownership	Reactor	Average Column Placement	Number of Quarters in Column 1	Quarters in ROP Column 1	Number of Quarters in Column 2	Number of Quarters in Column 3	Number of Quarters in Column 4	Number of Quarters in Column 5
Utility Owned	Arkansas Nuclear One Unit 1	1.14	49	87.5%	6	1	0	0
Utility Owned	Arkansas Nuclear One Unit 2	1.07	54	96.4%	0	2	0	0
Utility Owned	Grand Gulf	1.13	49	87.5%	7	0	0	0
Utility Owned	River Bend	1.14	48	85.7%	8	0	0	0
Utility Owned	Waterford	1.23	43	76.8%	13	0	0	0
Utility Owned	Utility Owned Average	1.17	47	83.3%	9	0	0	0
Wholesale Commodity Managed	Cooper	1.95	26	46.4%	15	7	8	0
Wholesale Commodity Owned	FitzPatrick	1.20	45	80.4%	11	0	0	0
Wholesale Commodity Owned	Indian Point Unit 2	1.63	38	67.9%	8	3	7	0
Wholesale Commodity Owned	Indian Point Unit 3	1.13	49	87.5%	7	0	0	0
Wholesale Commodity Owned	Palisades	1.39	38	67.9%	14	4	0	0
Wholesale Commodity Owned	Pilgrim	1.23	46	82.1%	7	3	0	0
Wholesale Commodity Owned	Vermont Yankee	1.25	45	80.4%	8	3	0	0
Wholesale Commodity Owned	Wholesale Commodity Owned Average	1.30	44	77.7%	9	2	1	0

(b)(4), (b)(7)(D)

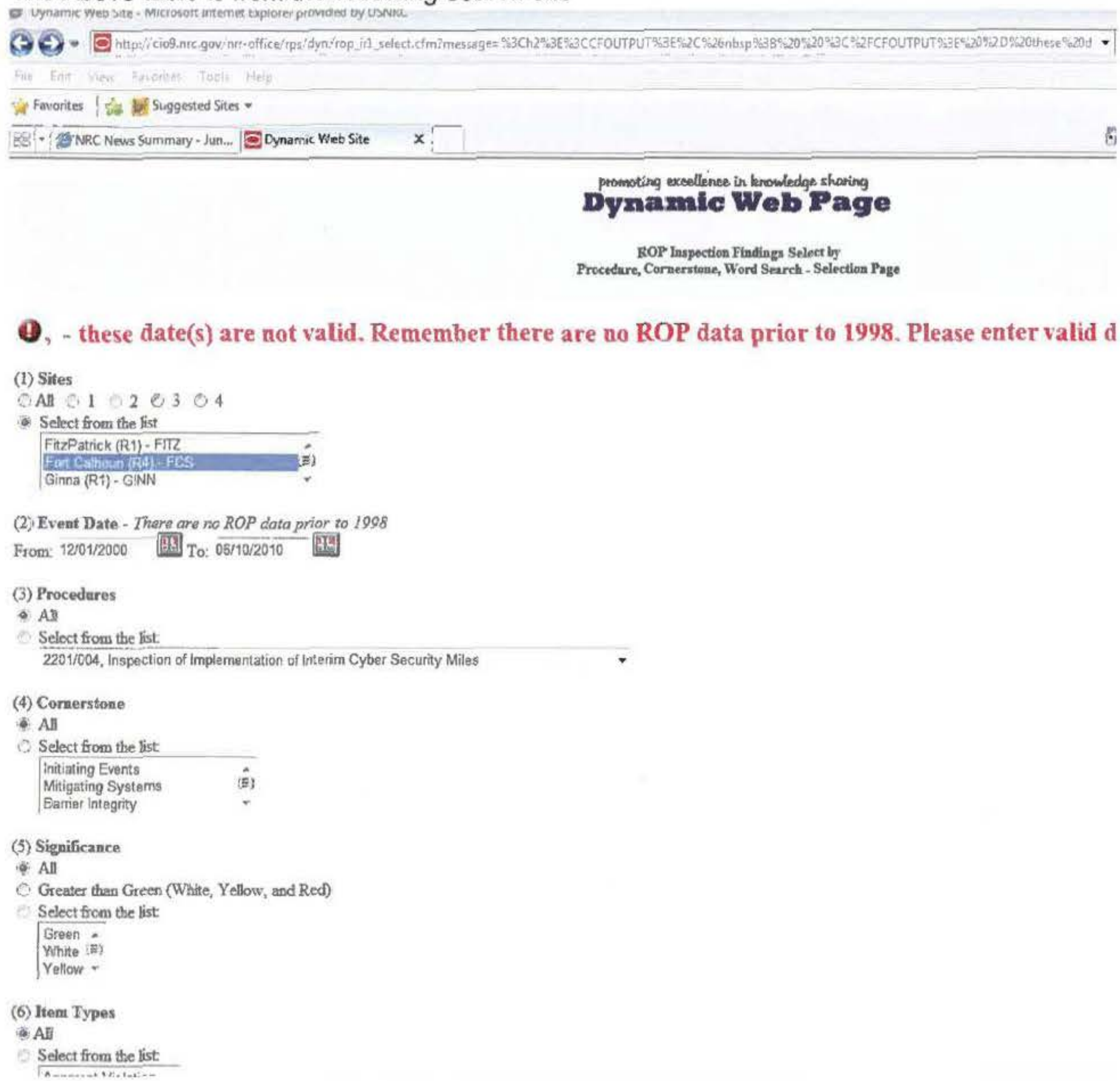
	ROP	LER	[redacted]
2000-2004	Finding - Green 11 Finding - N/A 6 Non-Cited Violation - Green 50 Non-Cited Violation - N/A 3 Violation - Green 1 Violation - SL-IV 1 Violation - White 1	14	
2005-2010	Finding - Green 8 Finding - N/A 3 Non-Cited Violation - Green 101 Non-Cited Violation - SL-IV 8 Violation - Green 1 Violation - SL-IV 1 Violation - White 4 Violation - Yellow 1	30	
2011-2015	Finding - Green 11 Finding - N/A 1 Non-Cited Violation - Green 174 Non-Cited Violation - N/A 8 Non-Cited Violation - SL-IV 1 Violation - Green 7 Violation - N/A 12 Violation - Red 1 Violation - White 6	58	

Percent of Time FCS spent in certain column.

Time Period	C1	C2	C3	C4	350
2000-Q1 2010 %	57.89474	34.21053	7.894737	0	0
2000-CAL Q3 2010 %	55	32.5	12.5		

ALL	Initiating Event	Mitigating System
Finding - Green 15 Finding - N/A 6 Non-Cited Violation - Green 121 Non-Cited Violation - N/A 1 Non-Cited Violation - SL-IV 4 Violation - Green 2 Violation - SL-IV 2 Violation - White 5	Finding - Green 3 Non-Cited Violation - Green 13 Non-Cited Violation - SL-IV 1	Finding - Green 8 Finding - N/A 1 Non-Cited Violation - Green 67 Non-Cited Violation - SL-IV 2 Violation - Green 2 Violation - White 4
Barrier Integrity	**Emergency Prep**	**Occupational Radiation Safety**
Finding - Green 1 Non-Cited Violation - Green 10 Non-Cited Violation - N/A 1	Non-Cited Violation - Green 1	Finding - Green 2 Non-Cited Violation - Green 24 Violation - SL-IV 2
Public Radiation Safety	**Physical Protection**	**MISC**
Non-Cited Violation - Green 5 Violation - White 1	Finding - N/A 1 Non-Cited Violation - Green 1	Finding - Green 1 Finding - N/A 4 Non-Cited Violation - SL-IV 1

The Above table is from the following search site


Dynamic Web Site - Microsoft Internet Explorer provided by USNRC

http://cio9.nrc.gov/nrr-office/rps/dyn/rop_ir1_select.cfm?message=%3Ch2%3E%3CCFOUTPUT%3E%2C%26nbsp%3B%20%20%3C%2FCFOUTPUT%3E%20%2D%20these%20d

File Edit View Favorites Tools Help

Favorites | Suggested Sites

NRC News Summary - Jun... | Dynamic Web Site

promoting excellence in knowledge sharing

Dynamic Web Page

ROP Inspection Findings Select by
Procedure, Cornerstone, Word Search - Selection Page

, - these date(s) are not valid. Remember there are no ROP data prior to 1998. Please enter valid d

(1) Sites

All 1 2 3 4

Select from the list

FitzPatrick (R1) - FITZ
Fort Calhoun (R4) - FCS
Ginna (R1) - GINN

(2) Event Date - *There are no ROP data prior to 1998*

From: 12/01/2000 To: 06/10/2010

(3) Procedures

All

Select from the list:

2201/004, Inspection of Implementation of Interim Cyber Security Miles

(4) Cornerstone

All

Select from the list:

Initiating Events
Mitigating Systems
Barrier Integrity

(5) Significance

All

Greater than Green (White, Yellow, and Red)

Select from the list:

Green
White
Yellow

(6) Item Types

All

Select from the list:

Sensitive

Initiating Event (Green 2011-2014)		
Finding	**Non-Cited**	**Violation**
	Component cooling water system for temporary off normal system conditions ***	
	Ensure leak before break commitment ****	
	Loss of reactor coolant *****	
	Loose maintenance carts, improperly storred ladders, excessive transient combustible material, scaffolding, removal of foreign material ****	
	5 alarm response procedure ****	
	480 VAC breaker *	
	Roving fire watch *****	
	HPSI pump flow imbalance (changing load on the main turbine) *****	

Mitigating Systems (Green 2011-2014)		
Finding	**Non-Cited**	**Violation**
CB20, panel A18, Window C3	Reactor protection system channel A trip unit 6 *	Diesel generator ****
Failure to generate complete inspection list	Safety injection refueling water tank vortex eliminator ***	Raw water strainer component ***
River sluice gates	Auxiliary feedwater ring *****	Auxiliary feedwater pumps FW-6 and FW 10 ***
Raw water pump anchor bolts	Reactor coolant pumps oil collection system	Fuel oil consumption calculations ***
Extent evaluation of emergency and abnormal operations procedure *	Scaffolding procedure *****	Raw water pumps ***
Frazil ice monitor	Power supplies *****	
Classifying component failures	test for CCW heat exchangers	
Raw water pump AC-10C	Vendor manual design control information ***	

	Adequate trending program ****	
	Electrical supply cable insulation for ccw motors *	
	Single worst cast spurious actuations	
	Alternate shutdown capability *****	
	post-fire safe shutdown procedure *	
	Safety injection tank *****	
	7 examples of procedure issues 6/7-2012 ****	
	Facility staff qualifications *****	
	Equipment available to measure river level locally ***	
	Design bases documents ****	
	Written evaluations for two changes for flooding mitigation strategies **	
	AFW pumps discharge check valve leakage *	
	Deficient evaluation of NRC bulletin 88-04 ***	
	Degraded/nonconforming condition evaluation and operability determination process *	
	CCW pump AC-3A *	
	Safety related pumps and valves up to code	
	Emergency feedwater tank FW-19 ***	
	Store raw water to emergency feedwater storage tank fill hose ****	
	CCW leakage criteria ****	
	Auxiliartt feedwater back pressure protection trip ***	
	Containment air coolers VA-16A and VA-16B ***	
	Raw water pipe support RWS-117	
	Raw water piping and piping support calculations *	
	Pipe supports SIH-17, SIH-94,	

	and SIH-12 ***	
	480V and 4160 V buss switchgear cabinets ***	
	Raw water pump anchor bolt ***	
	18 alarm response procedures *	
	Emergency operating procedure *****	
	Air operated valve elastomers *	
	Failure to update calculations to account for non-safety related loads supplied by the EDG***	
	Fuel oil in tank FO-1 into applicable design documentation ***	
	Fuel oil inventory calculation ***	
	Raw water flow into intake structure ***	
	Implement maintenance rule	
	120 Vac system *	
	Monitor performance of penetration seals	
	Restoring temporary modifications ****	
	Digital low resistance ohmmeter values ***	
	6 instances of failure to identify a deficiency or a condition adverse to quality and enter them into CAP *	
	11 instance of failure to initiate condition reports *(****)	
	Loss of raw water *****	
	Failure to identify significant condition 8	
	Low river level ***	
	Risk-based operability determinations ****	
	Operability determinations that lacked adequate technical justification ****	
	Intake cell level sluice gate leakage ***	
	480 Vac 1B4A bus breakers *	
	Structural calculations related to RCS ***	

	Overstressed components ****	
	Turbine driven auxiliary feedwater pump FW 10 ***	
	Operability determination process ****	
	480 V breakers quality assurance records	
	Switchgear room cooling ***	
	Station procedure FCSG-24-4 **** (2)	
	Fire protection program	
	Sluice gate leakage valves ***	
	Piping in the intake structure raw water vault*	
	MSPI SSFF degrading trend *	
	Removal of motor for raw water pump B on the intake cell level control during a potential site flood ****	
	Procedure for intake cell level control ****	
	Class 1 raw water piping in non-class 1 service building ****	
	Piping and pipe supports ****	
	Evaluate safety impact of degraded conditions ****	
	Raw water piping supports ***	
	Containment air cooler pipe supports VAS-1 & VAS-2 ***	
	HPSI injection valve ***	
	Failure to request a license amendment for a required change **	
	HPSI pump design and runout***	
	CS design change **	
	Operability procedure ****	
	Class 1 structures wall thickness defiiciences *	
	Raw water electrical pull boxes PB-128T and PB-129T	
	Reactor Vessel Head structural elements ***	
	Containment internal structure and auxiliary building *	
	Operability determination procedure ****	
	PMT procedure ****	
	Operability determination ****	

	Model flow path for external flood mitigation ***	
	Raw water pump 10C **	
	DG starting air system *	
	Software classification issues *	
	Switchgear room cooling ***	
	Failure to initiate condition reports for gaps identified in resolving NRC non-cited violations****	
	Equipment not in state of readiness	
	Raw water strainer AC-12B ***	
	Refill CCW system **	
	Gas voiding of ccw *	
	Control panel of raw water strainer AC-12B	
	Valves in auxiliary feedwater system *	
	Safety-related pipe stress calculations ***	
	Evaluate operability of degraded or non-conforming conditions ****	
	CS system surveillance test	
	Design of FO-10 to FO-1 fuel oil transfer system ***	
	Temperature limits in auxiliary building	
	Installation of flood barriers caused potential inoperability of auxiliary feedwater system ***	
	Internal flood analysis *	
	Adverse design changes ***	
	Thermal lag analysis ***	
	Non-conservative values ***	
	Failed to monitor HCV-2875A	

Barrier Integrity (Green 2011-2014)		
Finding	**Non-Cited**	**Violation**
	Decontamination work in spent fuel pool canal *****	Containment pray runout *
	HE-2 crane	

	Containment spray system ***	
	Reactor coolant pumps OI-RC-9,annunciator response procedure, control element assembly, reactor coolant system high activity ****	
	Spent fuel pool area charcoal filtration system VA-66 *****	
	Containment internal structures ****	
	Containment electrical penetration assemblies ***	
	Workers failure to follow work control procedures *****	

Criterion	**Symbol**
Corrective Action	*
50.59	**
Design control	***
10 CFR part 50 Apendix B, Criterion V	****
Technical Specifications	*****
USAR	******

Initiating Event (Green 2000-2010)		
Finding	**Non-Cited**	**Violation**
Inadequate Operator Control during low power ops	Socket weld on discharge piping for the charging *	
Ineffective corrective actions for hydrazine spills *	Reactor coolant system parameters ****	
LCV 1190 condensate control valve *	Venting the reactor vessel head ****	
	Bus bars due to high wind *****	
	Transient combustion limit in room 59 *****	
	161 kilovolt power to a safety related busses *****	
	Inadequate internal flooding procedure from pipe break *****	
	Inadequate procedure for plant cooldown (2700 gallon steam void in rcs) *****	
	Re-pack pressurizer spray valve PCV 103-1 *****	
	Failure to perfrom risk assessment in vicinity of T1 transformer	
	Inadequate maintenance on OP-PM-FP-1000 (fire protection system flushing) *****	

Mitigating Systems (Green 2000-2010)		
Finding	**Non-Cited**	**Violation**
Intrument Air Check Valve IA-HCV-386-C	Reactor coolant gas system	Diesel generators ****
480 volt Motor Control Center MCC-3C2 (portable heater loads	Fire area 32 ******	Raw water strainer components design basis ***
Low pressure safety injection	Inadvertent manual start of	

system header	diesel driven auxiliary feedwater pump	
Fire protection sprinklers	CCW inlet isolation valves *	
Fuel oil inventory not being verified	Nonlaod shed welding receptacles *	
Containment tendon stressing gallery door *	Failure to update surveillance procedure following valve configuration changes	
Nonconservative controls of containment cleanliness	Failure to obtain shift manager approval *****emergency diesel generator fuel oil inventory *****	
	Auxiliary feedwater pump **	
	Emergency diesel generator air starting system air relay valves *	
	Auxiliary feedwater pumps	
Diesel driven auxiliary feedwater pump coupling guard	Containment tendon stressing gallery work *****	
Raw water/CCW heat exchangers AC-1A, A-1B	Containment tendon stressing gallery work *****	
Raw water system	Drum heater plugs *****	
Diversion of internal flood water to ECCS pump rooms	Nonload shedding electrical outlets *	
	Control room air conditioner ****	
	Safety related 4 kV bus ground detection circuitry ****	
	Train of charging pump	
	Frazil ice buildup *****	
	Diesel generator test procedure ****	
	Diesel generator test procedure *****	
	Containtment piping penetrations M-9 and M-12	
	Long-term loss of instrument air	
	RAW water pump AC-10B	
	Long term loss of instrument air***	
	Inadequate tech spec 2.4 ****	
	Room 62 and 69 (fire barriers) *****	

	Fire protection to features for components important for cold shutdown	
	Fire hose station FP-7G *****	
	Failure to follow documented instructions *****	
	Potential compromise of scenario requalification examinations	
	Fire protection program implementation btwn rooms 1 and 58 and rooms 1 and 30 *****	
	Containment protective coatings inspection ****	
	Turbine driven auxiliary feedwater pump***	
	Loss of raw water *****	
	Fire water as a backup for raw water	
	Fire protection program *	
	Reactor coolant pump seal O-rings	
	Low pressure safety injection water hammer ****	
	Fire protection program(fire door 1007-10 between fire area 20.1 and fire area 20.4 *****	
	Station fire plan, revision 61, attachment 14 *****	
	Decay heat removal loops *****	
	Alternate shutdown procedure *****	
	CVCS and HPSI piping	
	Raw water pumps *****	
	Component cooling flow element *	
	Two high pressure safety injection pumps ***	
	Component isolation valve ***	
	Raw water strainer component ***	
	Loss of CCW *****	
	Turbine driven auxiliary feedwater pump *	
	Emergency diesel generator *****	
	Raw water pumps and	

	strainers *	
	Emergency diesel generator-1 fuel oil transfer pumps set points ***	
	Procedure OI-EW-1, Extreme weather revision 13 *****	
	Component cooling heat exchangers AC-1A-D CCW bypass line isolation valve ***	
	Redundant trains of auxiliary feedwater *****	
	Boron acid leaks *	
	Raw water pump AC-10D packing leakage *	
	Diesel generator control cabinets ***	
	Boric acid corrosion control procedure ****	
	Post fire safe shutdown procedure *	
	Fire water supply system piping *	
	Turbine building concrete floor *	
	Emergency diesel generator relays ****	
	HPSI header-alternate header isolation valve HCV-2987 *****	
	Raw water pumps ***	
	Raw water pump power cables *	
	Safe shutdown at flood level 1009.3 ****	
	Cold shut down impossible if river exceeds 1009 feet	
	Repeated tripping of turbine driven auxiliary feedwater pump FW-10*	
	Turbine driven auxiliary feedwater pump *****	
	Turbine driven auxiliary feedwater pump exhaust backpressure trip lever *****	
	Inadequate desing margin Engineering change 45105***	
	4160 V and 480V *****	
	Fuel oil pump FO-37 ***	
	Connection btwn cable lugs an cables *****	

	Apply code ASME section XI code case N-513-2 raw water pipe	
	Operating test given to licensed operators	

Barrier Integrity (Green 2000-2010)		
Finding	**Non-Cited**	**Violation**
Failure to check Main hoist limit switches	Spent fuel handling machine operator *****	
	Test backup nitrogen supply systems to ccw inlet and outlet valves to containment air cooling units	
	Failure to certify nondestructive testing personnel	
	Outside contractor properly qualified ****	
	Transferring fuel in reactor vessel *****	
	ASME code requirements for Air Accumulators	
	Personal access lock door equalizing valve leakage *	
	CCW valves to containment cooling units *****	
	CS pumps *	
	Auxiliary building crane operating instructions ****	
	FC 05561, CCW relief valve setpoints, PCS-212 and 413 ***	

Criterion	Symbol
Corrective Action	*
50.59	**
Design control	***
10 CFR part 50 Apendix B, Criterion V	****
Technical Specifications	*****
USAR	******

Inspection Report 2005005 (2/5/06)		
	Equipment	Findings
1R08 in-service inspection activities	N/A	N/A
1R12Maintenance rule implementation	• Emergency diesel generator 2 • Electrohydraulic control Pump EHC-4B	None
1R13Maintenance Risk Assessment and emergent work evaluation	• Raw water pump breaker AC-10A • Raw water.ccw Heat exchanger AC-1C • Air compressor CA-1A • Pressurizer level check surveillance • LPSI header • Condensate makeup control valve LCV-1190 backup nitrogen supply bottles • Motor driven Fir Pump FP-1A • LPSI piping • Raw water pump AC-10D	none
1R15 Operability evaluations	• TCV-202, RCS loop 2A temperature control valve • FT-1396, Steam generator RC-2A inlet flow transmitter • RM-063, accident range stack gas radiation monitor remote rate meter • 161 kV grid voltage • LPSI system	Non cited violation for LPSI (green)
1R17 50.59	N/A	N/A
1R19 post maintenance tests	• Replace regulator gage on IA-HCV-489B-FR, HCV-489 instrument air supply filter/regulator • Install diesel generator 2 jacket water filter skid • Remove PI-325-1 • Replace YS-351 • Lube SI-2B coupling,	None

	change oil, and obtain oil sample • Calibrate PI-323B HPSI pump SI-2B discharge pressure	
1R21 CDBI	N/A	N/A
1R22 Surveillance Testing	• Channel A safety injection, containment spray, and recirculation actuation signal test • Fire protection system • Component cooling valves C & D	None
1R23 Temporary modifications	EC36650, reverse the logic of reactor vessel flange leakoff indication Pressure Switch PS-139	None

Inspection Report 2005004 (11/14/05)		
	Equipment	Findings
1R08	N/A	N/A
1R12	• Engineering safety features system switch CS-A/LS • Loop 2 to shutdown cooling isolation valve HCV-348	None
1R13	• Safety injection loop injector valves HCV-331 & HCV 333 • Charging pump 1C • EDG 1 & 2 • Auxiliary feedwater pumps FW-10 and FW-6	none
1R15	• Condensate makeup control valve LCV-1190 backup nitrogen supply • Containment cooling and filter unit VA-15A cooling coil VA-1A • Pressurizer RC-4 relief isolation valve HCV-151 • Auxiliary feedwater	LCV and PUMP FW-54 (TBD)

	pump FW-10 back pressure trip latch FW-64	
1R17	N/A	N/A
1R19	• SI-123(SI-1B drain) • VA-46B hot gas valve • Condenser fan motor for VA-46B • AC-10D pump	none
1R21	N/A	N/A
1R22	• Channel B safety injection, Containment spray and recirculation actuation signal test • RCS leak rate • Pressurizer level instrument L-101X &L-101Y • DG1 • Raw water instrument air accumulator check valve • AC-10D Raw water pump	None
1R23	Turbine EHC master trip solenoid valves	None

Inspection Report 2005002 (5/12/05)		
	Equipment	Findings
1R08	N/A	N/A
1R12	• Auxiliary building ventilating and Air conditioning system condensing units VA-95 &VA-86 • Intake Structure Sump Pumps, VD-2A & VD-2B	None
1R13	• Auxiliary Feedwater Pump FW-10 and main feedwater pump FW-4A • Containment spray pump SI-2A • LPSI pump SI-1A • EDG 1	none

	• Spent fuel pool cooling system • 345 Kv elevtrical supply lines	
1R15	• Auxiliary feedwater pump FW-10 • Lower portion of reactor vessel head • Safety injection tanks SI-6A-D • Fill/Drain Line Relief Valve to RCDT WD-1 • SI-22 • EDG 1	Non-cited green for auxiliary feedwater pump FW-10 and for safety-related coatings
1R17	N/A	N/A
1R19	• Rod RC-10-23 • Temporary spent fuel pool cooling system • Disconnect Switch DS-T1A-1 • IA-HCV-2603B • Safety injection tank SI-6A-D supply inboard isolation valve	None
1R21	N/A	N/A
1R22	• IA-YCV-1045-C • Main steam safety valves • Safety injection tank discharge check valves • RAS leakage to SIRWT • Oil storage tanks FO-1 & FO-10 • M-39 and M-53	None
1R23	Temporary spent fuel pool cooling system	None

Inspection Report 2005003 (8/5/05)		
	Equipment	Findings
1R08	• Pressurizer Lower	None

	girth Weld PRZ-SC-3-403 • Steam Generator A Feedwater Nozzle Weld 16-FW-2001/12 • RPV closure head welds PRVCH-CRD-CO-41&-41-2 • Trapeze strut 8-AC-2003/01-PR • Steam Generator A extension ring to Shell weld SG-1-4b • Shutdown cooling heat exchanger AC-4B • Steam Generator A lower head to extension ring weld SG-1-C-2 • Steam generator tube	
1R12	• Circulating water pump CW-1A • Heatless Air Dryer CA-12 • Air compressor CA-1B • Control Room air condition units • Reactor coolant pump seals • Circulating water pumps • Safety injection refueling water tank recirculation valves	None
1R13	• Component cooling heat exchanger AC-1D • CCW oulet valve HCV-482B solenoid • CA-7 air compressor • FP-181 fire hose cabinet FP-4L hose connection valve • Blowdown tank FW-7 transfer pump FW-34B • LCV-1109 condensate	none

	makeup level control valve • EDG 2 • Auxiliary feedwater pump FW-54	
1R15	• Main feedwater pump FW-4A • Charging pump CH-1A • Control room ventilation system	none
1R17	ECCS containment sump screen	NONE
1R19	• Raw water pump AC-10C • Auto Load Shed Channel A control switch • CS geader isolation Valve HCV-344&345 • Control element assembly RC-10-41 • Vessel Seal Leakage instrument line waste drain line valve RC-163	none
1R21	N/A	N/A
1R22	• AFW injection check valves FW-163 &164 • RCS leak rate test • Auxiliary feedwater system flow transmitters	None
1R23	Fuel assembly AA06	None

Inspection Report 2005009 (12/7/05)		
	Equipment	Findings
4OA2	Inspectors reviewed 183 CR's over 2 year period	
1R08	N/A	N/A
1R12	N/A	None
1R13	N/A	none
1R15	N/A	

1R17	N/A	N/A
1R19	N/A	none
1R21	N/A	N/A
1R22	N/A	None
1R23	N/A	None

Inspection Report 2005011 (10/14/05)		
	Equipment	Findings
1R08	N/A	N/A
1R12	N/A	None
1R13	N/A	none
1R15	N/A	
1R17	N/A	N/A
1R19	N/A	none
1R21	• Raw water system • ASME code requirements Air Accumulaots • Intake structure	• NCV green for raw water system and ASME code • Green finding for intake structure
1R22	N/A	None
1R23	N/A	None

Inspection Report 2006002 (5/15/06)		
	Equipment	Findings
1R08	N/A	N/A
1R12	•	None
1R13	•	none
1R15	•	
1R17	N/A	N/A
1R19	•	none
1R21	N/A	N/A
1R22	•	None
1R23		None

Inspection Report 2005004 (11/14/05)		
	Equipment	Findings
1R08	N/A	N/A
1R12	• Auxiliary Feedwater Pump FW-54 • Potable water system	None
1R13	• CW-6A bearing water	none

	cooler • FW-3 condensate cooler • Main Feedwater Pump FW-4B • LPSI jockey pump SI-18	
1R15	• CS Pump SI-3B bearing cooler CW outlet flow controller • LPSI system • CCW system • Backup air instrument air to condensate makeup control valve	Unresolved item with backup instrument air
1R17	LPSI jockey pump	none
1R19	• FW-57 startup aux feedwater pump suction strainer • XSS control switch for CA-1C • Air compressor C • HCV-2851 • Component IA-DPT-1039-B1	none
1R21	N/A	N/A
1R22	• Channel A safety injection, containment spray and recirculation actuation signal test • Room 22 safety injection/containment spray pumps and valve • DG 1 • Safety injection and refueling water tank • Component cooling category B valve	None
1R23	Voltage recorder in control room	None

Inspection Report 2006003 (8/14/06)		
	Equipment	Findings
1R08	N/A	N/A
1R12	• 'A' circulating water cell • Coolant charging pump CH-1C	None
1R13	• EDG 1& 2 • Auxiliary feedwater FW-10 steam inlet valve YCV-1045 • CCW pump AC-3C breaker	none
1R15	• Containment spray header isolation valve • Air supply riser BK • Incorrect heat sink calculation • Reactor vessel monitoring system	none
1R17	N/A	N/A
1R19	• Containment purge isolation valve PCV-742A • Auxiliary feedwater pump • Raw water pump AC-10D backup seal water supply filter AC-22D • Raw water strainer AC-12B and backwash valve AC-2805B • Raw water pump AC-10B discharge valve	none
1R21	N/A	N/A
1R22	• Component cooling valves C & D • DG1 • AC-10B raw water pump • Fire protection system • Reactor manual trip	None

1R23	USAR (defeating annunciator card CB-20 A15 B3)	None

Inspection Report 2006004 (11/14/06)		
	Equipment	Findings
1R08	N/A	N/A
1R12	• Instrument air dryer failures • Fuel oil tank FO-38 level switch LS-2120	None
1R13	• Condensate storage tank • Review of licensee's risk assement • CCW pump AC-3B • 161kV off-side power	none
1R15	• DG2 • YCV-817B DG2 room fresh air supply damper • Containment duct relief port	
1R17	• Pressurizer • Steam generator large bore piping • Pressurizer heater cable replacement • Containment opning replacement SG • Reactor vessel head	NCV (green) pressurizer weight
1R19	• IA-HCV-2883B-FR • Charging pump CH-1A • SG RC-2A blow-down to blow-down tank FW-7 control valve HCV-1390 • Fire main rupture btwn FP-106 and FP-104 • HPSI SI-2C	none
1R21	N/A	N/A
1R22	• Spent fuel storage facility • SG RC-2B channel B	None

	pressure Loop B/P905 • RCS leak rate test • Safety channel C • Main steam safety valves	
1R23	N/A	None

Inspection Report 2006005 (2/14/07)		
	Equipment	Findings
1R08	• Class 1 welds • Pressurized water reactor vessel upper head penetration • Boric acid corrosion control • SG tube	none
1R12	• CS injection valve HCV-345 • CCW pump AC-3B	None
1R13	• 345 kV electrical supply • Auxiliary FW-54 • DG 1 • CCW pump AC-3C • Condenser off-gas radiation monitor replacement RM-057	none
1R15	• CCW nitrogen bottle regulators • CCW pump AC-3B breaker 1B4A-1 • CS injection valve HCV-345 • RCS flow intruments tubing separation • CH-143 &-155	2 NCV (green) dealing with CCW pump
1R17	N/A	N/A
1R19	• RCS leakage testing • Steam generator	none
1R21	N/A	N/A
1R22	• Steam generator • Type C local leak rate test of penetrations M-	None

	39 and 53 • CCW AC-3C	
1R23	N/A	None

Inspection Report 2007002 (5/15/07)		
	Equipment	Findings
1R08	N/A	none
1R12	• Containment sump strainer • Loss of shutdown cooling	None
1R13	• DG 1 • Auxiliary feedwater pump FW-6 • LPSI pump SI-1A	none
1R15	• Containment ventilation units VA-3 and VA-7 • Safety related 4160 kV breaker • Normal range stack gas radiation monitor remote ratemeter RM-062 • Auxiliary feedwater pump FW-6	none
1R17	N/A	N/A
1R19	• Filter regulator assembly • Boric acid tank CH-11B level indication transmitter loop L-254 • SG RC-2B auxiliary feedwater inlet valve limit swith HCV-1108-33A • SG RC-2A and RC-2B • Feedwater pump FW-4C recirculation valve FCV-1151C-20	none
1R21	N/A	N/A
1R22	• DG 2	None

	• Hydrogen purge test • Third auxiliary feedwater pump • Raw water system • Fire protection system •	
1R23	N/A	None

Inspection Report 2007003 (8/14/07)		
	Equipment	Findings
1R08	N/A	none
1R12	• RCS in-core instrumentation flange	None
1R13	• AC-10D raw water pump • HCV-329 • LPSI to loop 1A and HCV-333 • LPSI to loop 2B injection valves • Switchgear and turbine plant cooling water • Replace CW-300&301 raw water pumps AC-10B and 10C sparging valves • FC-2820 HPSI pump SI-2B bearing cooler CCW outlet flow controller • FW-54 diesel AFW pump • Air compressor CA-1C • CA-366 air dryer CA-12 pressure relief valve	none
1R15	• HCV-1749 containment service air header outboard isolation valve • Auxiliary feedwater pump FW-10 steam traps flow indicator FI-	none

	1138 • DG 2 • Reactor protective system	
1R17	N/A	N/A
1R19	• 480 V BRKR for AC-3C • AC-3C • QC-10A • AC-10D • HCV1749	none
1R21	N/A	N/A
1R22	• RCS flow rate determination • Channel B safety injection, CS and recirculation actuation signal test • AC-10A raw water pump • DG 1 • YCV-1045	None
1R23	N/A	None

Inspection Report 2007004 (11/13/07)		
	Equipment	Findings
1R08	N/A	none
1R12	• DG 1 • HCV-151 • FW-4B main feedwater pump	Green finding in 2007011 for DG 1
1R13	• DG 2 • Discharge valve FW-479 • HCV-554 • HCV-329 • Channel A safety injection, CS and recirculation actuation test • DG-1	none

1R15	• 161 kV and 345 kV lines • Auxiliary feedwater system • Ventilation Fan VA-64B • Ccw system • RCS • Station battery posts/terminals	none
1R17	N/A	N/A
1R19	• Emergency response facility computer system • 3CR auxiliary contacts on DG 2 • SA-194 primary starting air pressure regulation valve on DG 2 • Main feedwater bypass valve FCV-1105 • DG 1	none
1R21	N/A	N/A
1R22	• Raw water system • AC-10B raw water pump • Third auxiliary feedwater pump • DG 1	None
1R23	• Instrument panel AI-110	None

Inspection Report 2007005 (2/11/08)		
	Equipment	Findings
1R08	N/A	none
1R12	• CW-1A main circulating water pump • EDG and support systems • Raw water system • CS system	Green NCV for raw water pumps and strainers

	• Containment recirculation • Safety electrical distribution system • Safety-related structures • Corrective action program within maintenance rule program	
1R13	• EDG-2 • Air compressor CA-1B • CS pump SI-3A	none
1R15	• FW-6 and FW-10 • Reactor thermal limits • Underground diesel oil storage tank • Component cooling heat exchangers AC-1A-D CCW bypass line isolation valve	NCV green for CCW heat exchangers bypass isolation valve
1R17	DG 1 fuel oil pump molded case circuit breakers	NCV green for DG 1 fuel oil transfer pumps
1R19	• DG-1 • Air compressor CA-1B • CCW heat exchanger inlet valves HCV-489A/B • LPSI pump SI-1B • Auxiliary feedwater pump FW-6	none
1R21	N/A	N/A
1R22	• RCS leak rate test • CVCS pump check • Reactor protection system • Steam generator • O-ring seal	None
1R23	Steam generator-2a feedwater regulating bypass valve FCV-1105	None

Inspection Report 2007007 (9/7/07)		
	Equipment	Findings
1R08	N/A	none
1R12	N/A	None
1R13	N/A	none
1R15	N/A	none
1R17	N/A	N/A
1R19	N/A	none
1R21	• 4160 circuit breakers • Station batteries - including battery transfer switches • Nitrogen admission to component cooling water surge tank, pressure control Valve PCV-2610 • Component cooling water shutdown heat exchanger inlet Valve HCV-480 • Safety injection and refueling water storage tank level indicators • Raw water strainers • Air accumulators - outside containment • Safety injection pump room ventilation • Reactor coolant pump seal coolant heat exchangers • Turbine driven auxiliary feedwater governor • High pressure core injection minimum flow recirculation isolation Valves HCV-385 and -386. • Emergency diesel generator room ventilation • Safety injection refueling water tank discharge Valves HCV 383-1, 2, 3, and 4 • Safety injection recirculation - through	• NCV green for loss of CCW • NCV green for safety injection pump room 21 • NCV green for raw water system pump discharge strainers • NCV green for turbine driven AFW keep warm line bypass throttle valves MS-366 & 368 • NCV green component cooling surge tank nitrogen and demineralized water supply line isolation valves

	recirculation sump and safety injection refueling water tank • High pressure core injection pump - net positive suction head and sequencing of valve manipulation (including safety injection refueling water tank vortex calculation needed to evaluate pump net positive suction head) • High pressure core injection valve, motor-operated Valve HCV-312 • Raw water and component cooling water - interface • Containment spray system - isolation Valves HCV-344 and -345. • Control element assemblies • Low pressure core injection - jockey pump • Discharge side of component cooling water pressure control Switches 412 and 413	
1R22	N/A	None
1R23	N/A	None

Inspection Report 2007010 (10/26/07)		
	Equipment	Findings
4OA2	155 Condition reports reviewed	Two green NCVs
1R08	N/A	none
1R12	N/A	None
1R13	N/A	none
1R15	N/A	none
1R17	N/A	N/A
1R19	N/A	none
1R21	N/A	N/A

1R22	N/A	None
1R23	N/A	None

Inspection Report 2008002 (5/16/08)		
	Equipment	Findings
1R08	N/A	none
1R12	• Instrument Air compressor CA-1C	None
1R13	• Auxiliary feedwater pump FW-54 • Raw water pump AC-10A • Bearing water pump C-9A • Auxiliary feedwater pump FW-6 • Main feedwater pump FW-4B • Raw water pump AC-10D • Heatless air dryer CA-12 • Bearing water cooler CW-6A • Room 22 safety injection containment spray pump	none
1R15	• DG euxiliary contacts 2-CR • Offsite power low signal relays 27-74/T1A1 and 27-74/T1A2 • Charging pump suction relief valve CH-180 • Raw water pumps A/B/C (AC-10D)	Green noncited violation for raw water pump AC-10D
1R17	N/A	N/A
1R19	• Auxiliary feedwater pump FW-54 • DG 2 • Raw water pumps AC-	Green NCV for diesel fuel oil leak in DG-2

	10(B&C) • Pressurizer quench tank CH-223 • DG 2 Turbo Lube Oil Circulating Pump LO-40-2	
1R21	N/A	N/A
1R22	• ECCS pump • Third auxiliary feedwater pump • SI-1B LPSI pump discharge isolation (HCV-2938) & recirculation check valve (SI-304) • Safety related battery chargers • DG 1	None
1R23	N/A	None

Inspection Report 2008003 (8/12/08)		
	Equipment	Findings
1R08	• Reactor pressure vessel closure head • Pipe to elbow weld • Elbow to pipe weld • Valve to pipe weld and vice versa • Hot leg nozzle to safe end welds 22 & 28 • Cold leg nozzle to safe end welds 24,26,30,32 • Reactor vessel head to flange • Vessel upper head penetration (VT-2) • Boric acid corrosion control • Steam generator RC-2A&B	• NCV green for (boric acid)
1R12	• Raw water pump packing gland • RCP RC-3A	None

1R13	• Auxiliary feedwater pump • RCS • Refueling outage • Raw water pumps	none
1R15	• CVCS • Containment coolers • MSIV • Air coolers VA-3A/B and VA-7A/B and Dampers VA-56A/B and VA57-A/B	none
1R17	N/A	N/A
1R19	• FW-54 • Pressurizer • Control rods • HCV-1105 • Raw water pump AC-10D	none
1R21	N/A	N/A
1R22	• Main steam safety valve MS-279 • CS and LPSI pumps • HPSI pump • Refueling outage • 4160 volt breakers • Check valve SI-344	None
1R23	N/A	None

Inspection Report 2008004 (11/3/08)		
	Equipment	Findings
1R08	N/A	none
1R12	• Safety injection water recirculation tank suction valves LCV-383-1/2 • Raw water pump AC-10D • Instrument air compressor CA 1B	None
1R13	• HPSI pump SI-2A • CS pump SI-3A	none

	• Auxiliary feedwater pump FW-6 • HPSI SI-2B • T1A1 and T1A2 transformers • Raw water header piping	
1R15	• Jacket water temperature control valve JW-116 • EDG • ECCS train A suction piping • Fire protection piping	none
1R17	N/A	N/A
1R19	• CS pump • Control room air condition Unit VA-46B • DG-2 • FW-54 • Containment cooling unit VA-8B CCW outlet valve HCV-403C	none
1R21	N/A	N/A
1R22	• USAR • Fire pump FP-1A • Station batteries EE-8A/B • Auxiliary feedwater pump FW-10 • Relief valve tailpipe • Reactor coolant leak detection procedure • Raw water pump AC-10A	None
1R23	N/A	None

Inspection Report 2008005 (2/12/09)		
	Equipment	Findings
1R08	N/A	none
1R12	• Electro-hydraulic control system	None

	• Steam generator 'A' feedwater	
1R13	• Intake cell A • Inadvertent entry	none
1R15	• ECCS suction headers • SI-159/160 check valves • RCS vent to pressurizer quench tank HCV-180 • Auxiliary feedwater pump recirculation valve FCV-1369 • Redundant equipment on opposite trains	Green NCV (redundant systems)
1R17	N/A	N/A
1R19	• M-coil replacement in rod control cabinets • CCW pump AC-3A • Raw water system flow element FE-2890 • EDG-2 &1	None
1R21	N/A	N/A
1R22	• USAR • CCW pump AC-3A	None
1R23	N/A	None

Inspection Report 2009001 (11/13/09)		
	Equipment	Findings
1R08	N/A	none
1R12	• DG security • Containment sump outlet strainer SI-12A	None
1R13	• Condenser FW-1A hotwell level controller LC-1190 • Switchyard • Shutdown cooling heat exchanger AC-4A • DG-2	none
1R15	• Crane HE-2	Green HE-2 crane

	• FW-10 • DG-2 • HCV-509A and B • HCV-400A	
1R17	N/A	N/A
1R19	• Heat exchanger CCW inlet valve HCV-481 • Cooling coil VA-8B CCW inlet valve HCV-403A • Charging pump CH-1C • Crane HE-2 • Boric acid pump CH-4A	none
1R21	N/A	N/A
1R22	• USAR • DG-2 • Component cooling category B • AC-3C CCW pump • Component cooling category A	None
1R23	N/A	None

Inspection Report 2009002 (4/30/09)		
	Equipment	Findings
1R08	N/A	none
1R12	• Fire protection and drainage systems • Reactor protection system channel C	None
1R13	• FW-54 • FP-1B • AC-10C • Air compressor CA-1B • Bearing water cooling pump CW-6B • A charging pump CH-1A • FW-10	none

	• FW-2B	
1R15	• Seismic restraint SIS-185 • HCV-2893 • FW-10 • DG	none
1R17	N/A	N/A
1R19	• HCV-2880B • Raw water outlet valve 1A ccw/raw water exchange • FW-10 • AC-10B • HCV-2875A • FW-54	none
1R21	N/A	N/A
1R22	• USAR • FW-6 valve FCV-1368 • Check valves FW-173 and FW-174 • 13.8 kV emergency power • Raw water system	None
1R23	N/A	None

Inspection Report 2009003 (8/3/09)		
	Equipment	Findings
1R08	N/A	none
1R12	• DG-2 • FW-10	None
1R13	• FW-10 • CS pump SI-3A • DG-1 & 2 • FW-54	none
1R15	• FW-10 • Isolation Valve HCV-820B • Radiation Monitor RM050/51 • Raw water Heat Exchanger AC-1D • FW-6 (auxiliary)	none

	• Channel B reactor protection system • Valve MS-291 • Inverters A & B	
1R17	N/A	N/A
1R19	• FW-10 • Valve HCV-2880A • HCV-2851 • HCV-492A • Air Compressor CA-1B	none
1R21	N/A	N/A
1R22	• USAR • FW-10 • Fuel handling machine interlock • Raw water system • EDG • RCS	None
1R23	N/A	None

Inspection Report 2009005 (2/5/10)		
	Equipment	Findings
1R08	• ASME code • Vessel upper head penetration • Boric acid corrosion control • Steam generator	none
1R12	• Condensate pump FW-2B • Main generator output breaker	None
1R13	• Reactor vessel • Auxiliary building fire header	none
1R15	• DG-1 • Auxiliary feedwater pump FW-6 • LPSI pump SI-1A • Charging pumps	none
1R17	N/A	N/A
1R19	• Containment purge air	none

	inlet inboard isolation valve PCV-742C • Bus tie breaker • CS header isolation valve HCV-344 • Pressurizer power operated relief valve PCV-102-2 • Relay AI-31-TEST-PB-K2	
1R21	N/A	N/A
1R22	• USAR • Station batteries 1 and 2 • Channel A, safety injection, containment spray and recirculation actuation test • Raw water system • Raw water pump AC-10D • Penetration M-86 for valve SI-176	None
1R23	N/A	None

Inspection Report 2009006 (12/31/09)		
	Equipment	Findings
1R08	N/A	none
1R12	N/A	None
1R13	N/A	none
1R15	N/A	none
1R17	N/A	N/A
1R19	N/A	none
1R21	• EDG 1&2 • Containment Fan Cooler 7C & 7D • 125 Vdc Station Batteries 1 & 2 • Auxiliary feedwater pump FW6 breaker 1A3-16 • Raw water pump Breaker 1A3-9 • 161kV transform T1A-	• 2 NCV green for intake structure and 1 unresolved problem with intake structure

	3&-4 • CS pump breaker 1B4B-1 • Auxiliary feedwater pump FW-54 • Auxiliary feedwater pump FW-6 • Power operated relief valve 102-1 • CS pump 3B • Auxiliary feedwater steam admission valves AOV 1045 and 1045 A&B • Auxiliary feedwater containment isolation valve AOV 1107B&1108B • Intake structure	
1R22	N/A	None
1R23	N/A	None

Inspection Report 2010002 (5/12/10)		
	Equipment	Findings
1R08	N/A	none
1R12	• Main feedwater pump FW-4C • Safety related inverters EE-8H and EE-8J	None
1R13	• West raw water header outage • FW-54 • Floor plug removal above room 21	none
1R15	• Safety injection tank SI-6C leakage header • Heat exchange CH-7 backpressure control valve PCV-210 • Power operated relief and pressurizer safety valves tailpipe • FW-10	none

	• FW-6	
1R17	N/A	N/A
1R19	• Raw ater pump discharge header isolation valve HCV-2874B • B-reactor protective system • DG-1 • CS pump SI-3A • DG-2	none
1R21	N/A	N/A
1R22	• UFSAR • RCS • FW-10 • FW-6 • HPSI SI-2B	None
1R23	N/A	None

Inspection Report 2010003 (7/26/10)		
	Equipment	Findings
1R08	N/A	none
1R12	• Pump CW-1A • FW-54 • Fuel oil transfer pump FO-37	None
1R13	• 480 volt ground on bus 1B3A • CCW isolation valve HCV-2893 • M2 contactor	none
1R15	• PCV-102-1&2 • DG-1 • MCC-3A1 • FE-142 • AC-3A • AI-3-M2	none
1R17	N/A	N/A
1R19	• MCC-3A1 • FE-142 • HCV-317, breaker	none

	• AI-3-M2 • CH-10 inlet valve	
1R21	N/A	N/A
1R22	• USAR • DG-1 • raw water pump • FW-10 • DG-2 • Room 22 safety injection/containment spray pumps and valve	None
1R23	N/A	None

Inspection Report 2010004 (11/8/10)		
	Equipment	Findings
1R08	N/A	none
1R12	• Heat drain tank level control valve LCV-1199 • Breaker 1A31	None
1R13	• 345 kV to 161 kV transformer T3 • LPSI SI-1A • CS pump SI-3A • DG-1 • DG-2 • Air instrument air compressors	none
1R15	• Control room air conditioning unit VA-46B • Room 81 • FO-10 to FO-1	NCV green for storage tank FO-10 to FO-1
1R17	6 samples of evaluations; 15 samples of changes, tests, and experiments	NCV green for pressure control switches PCS-412&413
1R19	• Control room air condition unit VA-46B • Reactor protection system • Air compressor CA-1c	none

	• HPSI pump SI-2B discharge valve • HCV-2908 • LPSI pump SI-1B discharge valve	
1R21	N/A	N/A
1R22	• USAR • Third auxiliary feedwater pump • Radial peaking factor • RCS valve • FW-6	None
1R23	N/A	None

Inspection Report 2008002 (5/16/08)		
	Equipment	Findings
1R08	N/A	none
1R12	• Condenser off-gas radiation monitor RM-057 • FCS cycle 24 maintenance rule	None
1R13	• T1 transformer • CS pump SI-3B	NCV green for T1 transformer
1R15	• Main feedwater pump FW-4B suction piping • FO-1 • Ccw heat exchanger ac-1C endbell • HCV-321 • East raw water header	NCV green for raw water piping
1R17	N/A	N/A
1R19	• HCV-1150A • HCV-344&345 • LPSI pump SI-1A • Raw water pump AC-10B • CS pump SI-3B	none
1R21	N/A	N/A
1R22	• USAR • SG low pressure trip unit	Unresolved item SG low pressure

	• Channel A safety injection • Room 22 • Reactor coolant col and hot leg temperature loops	
1R23	N/A	None

2000	2001	2002	2003
Finding - Green 4 Finding - N/A 3 Non-Cited Violation - Green 7 Non-Cited Violation - N/A 2	Finding - N/A 1 Non-Cited Violation - Green 7 Non-Cited Violation - N/A 1	Finding - Green 5 Finding - N/A 1 Non-Cited Violation - Green 13 Violation - White 1	Finding - N/A 1 Non-Cited Violation - Green 11 Violation - SL-IV 1
2004	**2005**	**2006**	**2007**
Finding - Green 2 Non-Cited Violation - Green 12 Violation - Green 1	Finding - Green 3 Finding - N/A 1 Non-Cited Violation - Green 18 Violation - White 1	Finding - Green 1 Non-Cited Violation - Green 17 Violation - White 1	Finding - Green 1 Non-Cited Violation - Green 16 Violation - SL-IV 1 Violation - White 2
2008	**2009**	**2010**	**2011**
Finding - Green 2 Finding - N/A 1 Non-Cited Violation - Green 17	Finding - Green 1 Finding - N/A 1 Non-Cited Violation - Green 9 Non-Cited Violation - SL-IV 3 Violation - Green 1	Non-Cited Violation - Green 24 Non-Cited Violation - SL-IV 5 Violation - Yellow 1	Non-Cited Violation - Green 15 Non-Cited Violation - SL-IV 1 Violation - White 2

Green	Non-Cited	Cited
Self-Revealing	58	4
Inspection	252	33
Corrective action	50	8
MISC	3	

White	Violation
Self-Revealing	2
Inspection	9
Corrective action	3
MISC	

Yellow	Violation
Self-Revealing	
Inspection	1
Corrective action	1
MISC	

Red	Violation
Self-Revealing	
Inspection	1
Corrective action	1
MISC	

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ROP PIM Reports - Event Dates: 01/01/2012 - 06/17/2015 - Generated on 06/17/15
By Types, Cornerstones, Event Dates, Sites
Significance: All
203 *Open/Closed Final* items selected -
Site: 'FCS'

Finding - **Green**	11
Finding - **N/A**	1
Non-Cited Violation - **Green**	159
Non-Cited Violation - **N/A**	8
Violation - **Green**	7
Violation - **N/A**	12
Violation - **Red**	1
Violation - **White**	4

Cross Cutting Areas:

- SCWE - *Safety Conscious Work Environment*
- HP - *Human Performance*
- PIR - *Problem Identification and Resolution*

Finding						
Mitigating Systems	02/02/2012	FCS	Green	*SCWE: N	*HP: Y	*PIR: N
Docket/Status: 05000285 (C)						
Open: ML12079A224						
(PIM) Failure to Perform Extent of Condition Evaluation						
The NRC identified a finding for failure of the licensee to follow directions of an apparent cause evaluation to perform an extent of condition evaluation. Specifically, following the identification of an inadequate temporary design modification that rendered annunciator CB 20, Panel A18, Window C3 inoperable on July 5, 2011, engineering personnel failed to perform an extent of condition evaluation to identify other annunciator windows rendered inoperable by the design modification. The failure of engineering personnel to perform an extent of condition evaluation as directed by the apparent cause evaluation for a temporary modification following identification of an unexpected condition was a performance deficiency. The finding is more than minor because the failure to adequately implement the corrective actions associated with the temporary modification s identified deficiencies affects the equipment performance attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective of ensuring availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because it did not represent a loss of system safety function, did not represent the actual loss of safety function of a single train for greater than its technical specification allowed outage time, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the corrective action program component of the human performance area associated with work practices because engineering personnel failed to follow direction and ensure that an extent of condition review mandated by an apparent cause evaluation was performed.						
Mitigating Systems	11/17/2012	FCS	Green	*SCWE: N	*HP: Y	*PIR: N
Docket/Status: 05000285 (C)						

Open: ML12366A158						
(PIM) Failure to Properly Scope All the Pertinent External Flood Protection Features into the Walkdown List in Accordance with Industry Guidance NEI 12-07						
The inspectors identified a finding of very low safety significance (Green) for the licensee s failure to generate a complete inspection list, with all the external flood protection features credited in the current licensing basis documents for flooding events, to comply with NRC endorsed NEI 12-07, Guidelines for Performing Walkdowns of Plant Flood Protection Features. These walkdowns were being performed in response to a March 12, 2012, letter from the NRC to licensees, entitled, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident. Specifically, the scoping list did not include several active components, which are an essential part of Fort Calhoun s design basis flood mitigation strategy. The licensee entered the issue into the corrective action program and revised the scoping list accordingly. The performance deficiency was determined to be more than minor because it is associated with the Mitigating Systems Cornerstone attribute of Protection Against External Factors (Flood Hazard) and it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, in addition to not scoping the sluice gates into the Flooding Features Walkdown List, fourteen additional active components would not have been scoped into the walkdown list. This would have prevented the licensee from identifying that preventive maintenance tasks needed to be created, and some active components that are an essential part of the flood mitigating strategy would not have been inspected and tested. The finding was screened as very low safety significance (Green) because the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event. The inspectors determined the finding had a cross-cutting aspect in the area of human performance because licensee personnel did not properly apply human error prevention techniques such as peer checking and proper documentation of activities (H.4(a))						
Mitigating Systems	12/31/2012	FCS	**Green**	*SCWE: N	*HP: N	*PIR: Y
Docket/Status: 05000285 (C)						
Open: ML13045B055						
(PIM) Failure to Manage Functionality of the River Sluice Gates						
The team identified a finding for the failure to manage the functionality of the river sluice gates. Specifically, the licensee s preventive maintenance program requirements were not appropriately implemented for a period of 12 months and as a result, the functionality of the river sluice gates was improperly maintained. The team concluded that the failure to manage the functionality of the sluice gates was a performance deficiency that warranted further evaluation. Specifically, the licensee s preventive maintenance program requirements were not appropriately implemented for a period of 12 months and as a result, the functionality of the sluice gates was improperly maintained. The examples supporting this performance deficiency are as follows: 1) Failure to perform preventive maintenance and monthly testing on the river sluice gates for four months 2) Failure to perform monthly testing on two sluice gates on September 2012 3) Failure to perform monthly testing on all the sluice gates on October 2012 4) Failure to properly identify and timely enter conditions adverse to quality into the Corrective Action Program 5) Failure to demonstrate effective control of performance of the river sluice gates and to place the system in a monitoring program 6) Failure to make appropriate functionality assessment when the river sluice gates failed the monthly testing during August 2012 The licensee entered these issues into their corrective action program under numerous condition reports described in the body of this report. Using the guidance in IMC 0612, Power Reactor Inspection Reports, Appendix B, Issue Screening, the inspectors determined this finding affected the Mitigating Systems cornerstone. The finding is greater than minor because it is associated with both of the Mitigating Systems Cornerstone attributes of Equipment Performance and Protection Against External Factors and, it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, Significance Determination Process, and conducted a Phase 1 characterization and initial screening. Using Phase 1 Table 3, SDP Appendix Router, the inspectors answered						