Conte, Richard

Tim clock
NRR

From: Ennis, Rick
Sent: Tuesday, April 27, 2010 11:40 AM
To: Conte, Richard
Cc: Elliott, Robert; OHara, Timothy; Tsao, John; Lupold, Timothy; Manoly, Kamal; Burritt, Arthur; Cahill, Christopher; Schmidt, Wayne; Chernoff, Harold; Schulten, Carl; Cline, Leonard; Schroeder, Daniel; Balian, Harry; Honcharik, Michelle; Bowman, Eric; Miller, Barry
Subject: Salem AFW Piping Testing

Rich,

As follow-up to our discussion this morning regarding the Salem AFW piping pressure tests required by IWA-5244 and Salem surveillance requirement (SR) 4.0.5, I did some research on the NRC staff position related to whether the missed surveillance provisions of Salem SR 4.0.3 are applicable to surveillances which have never been performed (i.e., versus surveillances that were "missed").

The Pilgrim TIA dated 1/23/09 (ML083660174) states that "the NRC staff's position is that a missed SR is different than an SR that was never performed." Some of the key points in the TIA supporting this position are as follows:

1) Use of the word "frequency" [in SR 4.0.3] establishes an interval, a period of time, that includes an initial performance of the SR, and a specified time period to re-perform the SR thereafter, i.e., to repeat the surveillance.

2) SRs are performed at frequencies that are more often than the mean-time to failure of particular systems. Thus, *most SRs confirm that SSCs are operable given an operable finding at the previous testing interval.*

On 2/24/09 a public meeting was held between the NRC staff and the industry Technical Specification Task Force (TSTF). As discussed in the meeting summary dated 3/24/09 (ML090700535):

"The TSTF began a discussion of SR 3.0.3 [SR 3.0.3 for Standard Technical Specifications (STS) is same as SR 4.0.3 for Salem] and stated that a SR that has never been performed should be treated like a missed SR. The staff stated that a missed SR is not the same as a never performed SR, therefore SR 3.0.3 can not be applied to a never performed SR. The TSTF stated that it does not agree with a December 2008 TIA on the subject. The TSTF stated that a TIA from 1992 conflicts with the December 2008 TIA. The staff requested that the TSTF forward a copy of the 1992 TIA to NRC. The TSTF stated that licensees must state why they feel the system will pass a SR in order to ask for an SR 3.0.3 extension for a portion of a system that has never been tested. The staff agreed with the TSTF that a framework for treatment of "never performed SRs" could be developed. The staff stated its belief that this approach was the best way to resolve the differences in position between the staff and the industry on this topic."

By letter dated 5/1/09 (ML090230254), the NRC staff did not accept for review an industry proposal (TSTF-512) that would approve a change to the STS. The change proposed by the TSTF would have revised the STS to establish a new position interpreting surveillances that never were performed as equivalent to surveillances whose test intervals are inadvertently exceeded.

In subsequent discussions with the NRC staff, the TSTF indicated that TSTF-512 would be resubmitted to the NRC providing additional justification for its position. I talked to Carl Schulten in NRR's Tech Spec Branch and he confirmed that the TSTF has not submitted a revised proposal. In addition, Carl confirmed that the current NRC staff position is as stated in the Pilgrim TIA.

Bottom line, PSEG's use of SR 4.0.3 to justify a delay in performing a surveillance that has never been performed is contrary to our current interpretation on use of SR 4.0.3.

F/6

Please let me know if you have any questions.

thanks,

Rick
301-415-1420

-----Original Message-----
From: Conte, Richard
Sent: Monday, April 26, 2010 5:11 PM
To: OHara, Timothy; Tsao, John; Lupold, Timothy; Manoly, Kamal; Burritt, Arthur; Cahill, Christopher; Schmidt, Wayne
Cc: Ennis, Rick; Elliott, Robert
Subject: Need for conference call RE: FEA of Degraded Salem Unit 1 AFW Piping

we are looking to do a conference call on Wednesday at 300pm or 330 NLT 400pm to go over what we know about the number of documents that have come in. we think Unit 1 can safely startup in light of repairs and code compliance.

Hdqtrs is reviewing the FEA that will be used to support at Unit 1 past operability determination and root cuase report. not sure when the later two documents will be in but they are not needed for Unit 1 starup.

There is a tech eval on reduced rated pressure to 1275 that was reviewed also in order to support the past operability review. Not sure how it applies to Unit 2.

Unit 2 current operability and risk assessment (with 1.25 year exposure time on risk) is in on draft and we plan to engage licensee representatives tomorrow on what information supports the Jan 21, 2010 start fo the 1.25 years to the outage next spring in 2011.

Bottom line is looks like back in the construction days, Unit 2 was properly coated but Unit 1 was not. No definitive answers yet as to why, based on desing or documented as left or as found condition back in the 1970s.

We are also trying to deal with the acceptablilty of the Unit 2 operability determination based on an ASME pressure test that was never done and operational information that support flow measurements but may not be considered the alternate ASME unabated flow test per the same code.

With respect to the previous paragraph, a TIA on Pilgirm (ml 083660174) from ITSB seems to accept, partially, an industry position that the test can be deferred if there is a basis that the test will pass - still a violation for which we could issue NCV is green (preferred) or exercise enforcement discretion (least preferred since they were caught on this issue). Not sure the flow information (not test) is as sensitive as the pressure drop but then again the coating issue seems to be different from Unit 1. I need to talk to someone in TS branch and or Lupold on this issue, perhaps tomorrow before the conference call - what is a reasonable expectation that the pressure drop test will pass in the spring of next year? When we get a less draft oper det. we can forward it.

-----Original Message-----
From: OHara, Timothy
Sent: Monday, April 26, 2010 4:47 PM
To: Conte, Richard
Subject: FW: FEA of Degraded Salem Unit 1 AFW Piping
Importance: High

Rich,

Tim Lupold has asked John Tsao to forward the FEA to Kamal Manoly for review.

-----Original Message-----
From: Tsao, John
Sent: Monday, April 26, 2010 4:15 PM
To: Manoly, Kamal
Cc: Lupold, Timothy; OHara, Timothy
Subject: FW: FEA of Degraded Salem Unit 1 AFW Piping
Importance: High

Kamal,

Tim O'Hara of Region I forwarded me the FEA report for the Salem buried AFW piping. Tim Lupold asked me to forward the FEA report to you (see the first attached file). Attachment No. 2 is my assessment of the FEA report that I sent to Tim O'Hara this morning. Attachments No. 3 and 4 are the preliminary information for the FEA report.

Thanks.

John

-----Original Message-----
From: OHara, Timothy
Sent: Friday, April 23, 2010 2:23 PM
To: Tsao, John
Cc: Lupold, Timothy; Conte, Richard; Gray, Harold; Burritt, Arthur; Schroeder, Daniel; Balian, Harry; Cline, Leonard; Sanders, Carleen; Ennis, Rick
Subject: FEA of Degraded Salem Unit 1 AFW Piping
Importance: High

Hello John,

Here is the FEA we've been discussing. Note that PSEG is still reviewing but they have provided this copy which will most likely not change. Please review this and let us know what you think. Thanks.

Tim OHara

-----Original Message-----
From: Berrick, Howard G. [mailto:Howard.Berrick@pseg.com]
Sent: Friday, April 23, 2010 2:11 PM
To: Schroeder, Daniel L.; OHara, Timothy
Subject: Evaluation of Degraded Underground Auxiliary Feedwater Piping (SIA Report 1000494_301_RC)
Importance: High

Attached ids the SIA Report RE: Evaluation of Degraded Underground Auxiliary Feedwater Piping

Please note: This report has not been through the PSEG Owners Acceptance or Third Party Review process.

Howard Berrick
PSEG Nuclear LLC

Salem Regulatory Assurance
PSEG Nuclear - Salem Generating Stations
(W) 856-339-1862
(Fax) 856-339-1448
(Bpr) (b)(6)

<<1000494_301_RC.doc>>

Sept 23, 2009
ml 092790463

APPLICABILITY

SURVEILLANCE REQUIREMENTS Lugold Taylor / Patrick

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be failure to meet the Limiting Condition for Operation, ~~except as provided in Specification 4.0.3~~. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

REACTOR COOLANT SYSTEM

3.4.11 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.11.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.11.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS ISI Prog.

4.4.11.1 In addition to ~~the requirements of Specification 4.0.5~~, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

4.4.11.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The No. 21 Steam Generator channel head shall be ultrasonically inspected in a selected area during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated manual activation switches in the control room and flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

d. LCO 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each non-automatic valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

2. Verify the manual maintenance valves in the flow path to each steam generator are locked open.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days on a STAGGERED TEST BASIS by:

1. Verify that the developed head of each motor driven pump at the flow test point is greater than or equal to the required developed head.

2. Verify that the developed head of the steam driven pump at the flow test point is greater than or equal to the required developed head when the steam generator pressure is >680 psig. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after secondary side pressure is greater than 680 psig.

c. At least once per 18 months by:

1. Verifying that each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

2. Verifying that each auxiliary feedwater pump starts automatically on an actual or simulated actuation signal.

The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump, provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 omponent(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant system temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

4.4.10.1.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The steam generator channel heads shall be ultrasonically inspected during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material. The stainless steel clad surfaces of the steam generator channel heads shall also be visually inspected during the above outages. This may be accomplished by direct visual examination or by remote means such as television camera. If the visual examination, either direct or remote, reveals detectable cladding indications, a record shall be made by means of a video tape recording or photographs for comparison purposes.