(6)

Rivera-Ortiz, Joel

From: Butcavage, Alexander -R II
Sent: Tuesday, September 20, 2011 2:39 PM
To: Rivera-Ortiz, Joel
Subject: RE: 11-520 - Summary Report of August 23/ 2011 Earthquake Response and Restart Readiness Determination Plan North Anna Power Station Units 1 and 2 and ISFSI

My 2 cents worth on North Anna....
After reading the internals portion OF THE North Anna Summary.....
I had to go back and read sections of the entire document....in order to make sense of Section 3.

1) The assessment appears to be based on an approach similar to SQUG approach to compare selected system walk down damage assessments derived from the guidance provided in EPRI NP-6695, and then projects that assessment to the RV internals.....

2) I do not have historic knowledge about the 1989 EPRI NP-6695 paper, but briefly reading portions yesterday, from an overall assessment of damage intensity I see where they are coming from. Sounds like a basis for the SQUG Program, and North Anna picked it to assess the damage.

3) Extrapolating from the 6695 paper to the internals appears to be a stretch..... just in the comparing the time frame of knowledge from 1989 to present day internals, and the OE we have over the last several years....i.e. D.C. Cook broken Baffle former bolts in one row across one baffle plate .....(Root Cause in progress when I left???? Not sure how it came out....or Again D.C. Cook, other unit, broken lower radial support bolts X-750 material??)
   a. They seem to make leaps of faith as we use to call them.....in several areas....the design functions of the internals.....can be maintained.....each plant is worded a little differently.....essentially it is to provide core support and maintain a coolable geometry
   b. They specifically say in paragraph 3.....For their evaluation , the criterion used for structural integrity is that "no dimensional changes occur (i.e. no yield in internals components) if this criterion is met, dimensions of the RV internals components are maintained. The RV Internals will thus continue to perform their design functions."

      My question is, they imply they are going to check measurements to ensure no reactor internals dimensional changes.....to insure structural integrity....did they actually do any measurement checks to support this?

      Then they go on to say structural integrity will be confirmed by application of a conservative criterion to calculated load results from either the operating basis earthquake (OBE or Design Basis Earthquake(DBE) to confirm margin exist. (What is this conservative criterion they refer to?)

   c. They go on to apply EPRI criteria from 6695. I would question the use of 6695 for internals...and ask did 6695 address internals at all? Why is it silent on internals if in fact it is? (Is it because this was a basis for the SQUG type approach for big picture damage assessment which, as I recall, was intended for application in cases where there was no vendor available to supply seismically qualified equipment? Not for seismically designed structures such as the reactor vessel and internals? (Assuming they are ASME plant here not B31.1 or .7)

4) At a minimum, I would be looking for:
Inside RV Issues:

F/24

a. Was there any delay in rod drops during the initial shut down? Were all poles in holes when they were supposed to be (1.7-2.4 seconds at Ginna as I recall) or did any hang up? (Plant computer history tie in???)
b. Any difficulty in retracting the Thimble tubes before refueling? Fuel Shifted?
c. Any fuel damage (leakers) or physical marks in grid strap area of fuel bundles or on baffle plates that would indicate that the fuel bundles contacted the baffle assembly. (Compare before and after chemistry for leakers)
d. Was there a change in the RV head Seal Leak off rate after the event?
e. Where there any fuel handling machine indexing issues when removing fuel from the RV during offload?
f. Any BFB heads (or other parts) on lower core plate, after fuel removed.
g. MRP-227 exam on upper flange weld if any indications of movement were noted. This could be done from inside surface if the core barrel is not removed. (Expand if any indications)
h. If core barrel pulled, scratch marks between upper flange of core barrel and flange of RV or head? Could see.... perhaps on head side surface when head is on the stand. If core barrel out, lower radial support damage on core barrel or RV wall sides. (Broken Bolts or impact marks) (not sure how you would do this without removing the core barrel to check for indications at support points, lower radials or flange support or any built in contact points for scratch marks.)
i. Does North Anna apply the reduced pass head stud tensioning program or the program that omits every other head stud out? IF so, did the analysis for those reduced head bolting programs (or any other programs that post date the EPRI 6695 work) consider the higher earthquake loads experienced by North Anna?
j. When putting fuel back in....do they need the shoe horn?
k. Does rod drop testing show deltas in drop times?
l. Any hang-ups when jacking thimbles back in?

Outside RV:

a. Damage to any of the thimble travel tubes in area below vessel if they are supported after bend radius...
b. Start with RCS Piping Analysis..... Any tell-tale signs of displaced primary equipment displacement. Look at sliding shoes on RV supports... i.e. is there a debris pile on one side of the lug attached to RV Nozzle....sign of displaced vessel.
c. Check gaps at all RCS support points. Get primary system gap set points from RCS piping analysis and field check existing gaps. Compare to analysis predictions. Are they per prediction or are they different. IF different then analysis predictions, explain why the delta's are acceptable.
d. Check for Boric Acid leaks "From above head" region of CRDM. Penetration Conoseal weld, latch housing omega seal or bolted joint, and the rod travel housing cap conoseal welds.
e. Check for impact at cap area of rod travel housing to seismic support plates at upper most part of the CRDM rod travel housing.....if vessel did shake, I would expect to see impacts marks on that seismic support plate or the stainless rod travel housing.....this in my mind is probably **one of the more important points** to check to ensure **no pressure boundary damage** that could be future leakage site. (Highest point of the CRDM pressure housing and sometimes there are gaps at the seismic restraint.
f. Did North Anna perform any MSIP at Alloy 600 weld joints or structural weld overlays/inlays. Do those calculations address the increase in seismic loadings? As I recall, MSIP was not applied until early or mid 1990's .....Therefore, I would question if any newer type analysis/modifications such as MSIP are covered by the EPRI 6695 work?
g. Does LBB basis include argument for loads combined from DBE and RCS break? Does higher earthquake load increase LBB analysis loads?
h. Are there any other post EPRI report time frame (1989) programs in place that were not addressed by the EPRI 6695 approach.
i. Look for containment penetration cracking....especially Main Steam and Feed-water lines if outriggers on supports are employed.

Flux Rate Monitored through NI's, check design of how the NI's were supported? Would that explain the Rate fluctuations in that the tubes that contain the NI detectors could have been shaking....causing perturbations in the core

axial and quadrant tilts? Thus rate of change sensed as a maximum and the set point was hit and caused rate monitor to trip Rx??

From: Rivera-Ortiz, Joel
Sent: Monday, September 19, 2011 10:09 AM
To: Butcavage, Alexander
Subject: FW: 11-520 - Summary Report of August 23/ 2011 Earthquake Response and Restart Readiness Determination Plan North Anna Power Station Units 1 and 2 and ISFSI

FYI

From: McCoy, Gerald
Sent: Monday, September 19, 2011 8:10 AM
To: Rivera-Ortiz, Joel; Munday, Joel; Christensen, Harold
Subject: FW: 11-520 - Summary Report of August 23/ 2011 Earthquake Response and Restart Readiness Determination Plan North Anna Power Station Units 1 and 2 and ISFSI

Joel,

Here is the report from Dominion. There is an enclosure on ISFSI analysis.

Gerry

Gerald J. McCoy
Chief, Reactor Projects Branch 5
Division of Reactor Projects, Region 2
United States Nuclear Regulatory Commission
Office: 404-997-4551
Cell: (b)(6)

From: Ginger L Rutherford [mailto:ginger.l.rutherford@dom.com]
Sent: Saturday, September 17, 2011 1:50 PM
To: Cotton, Karen; Davis (NMSS), Jennifer; Hiland, Patrick; Khanna, Meena; Manoly, Kamal; (b)(6) ; Mendiola, Anthony; Howe, Allen; Martin, Robert; McCoy, Gerald; Boyle, Patrick; Karwoski, Kenneth; Giitter, Joseph
Subject: 11-520 - Summary Report of August 23/ 2011 Earthquake Response and Restart Readiness Determination Plan North Anna Power Station Units 1 and 2 and ISFSI