From: Gary Demoss , RES
To: Mark Kirk
Date: 10/8/02 10:38AM
Subject: Re: Davis-Besse

Mark,

I'm not sure what your boss wants - but it would help me if I knew about what your comments will be on the current NRR package are. (i.e., no comment?, major comments? , can't do comment until November?, etc.)

Gary

H-124

Mid-October
Draft

Probabilistic Analysis & Lessons Learned Issues Associated the Davis Besse CRDM Nozzle Cracking and RPV Head Wastage

Gary DeMoss, Chris Hunter
RES/DRAA/OERAB
October xx, 2002

Status Report Based on Predecisional Draft SDP (October 2002) and Lessons Learned Task Report (October 2002)

AS-FOUND CAVITY RUPTURE - MLOCA

NRR Initiating event analysis - based on the rupture probability distribution multiplied by the frequency of pressure transients between operating pressure (2165 psi) and SRV set point (2500 psi)

- expected values of rupture pressure are around 6000 psi (various analyses).
- Probability of failure in last year $\leq 2 \times 10^{-4}$

Conditional Core Damage Probability given initiating event - SPAR Model and licensee provided values are about 3×10^{-3} for MLOCA

NRR-estimated ΔCDF - $\leq 6 \times 10^{-7}$

Issues/uncertainties - Operating on the tail of the rupture probability distribution; Current analysis does not consider flaws in cladding material or newly found cracks.

CRDM NOZZLE EJECTION- MLOCA

NRR Initiating event analysis- Used a probability of rupture vs. time curve from ANL.

- Assumed nozzle 3 wetted for 8 years ($P_{eject.} = 1.6 \times 10^{-2}$).
- Then assumed nozzle 2 wetted for 4 years and nozzle 5 wetted for 2 years. They added 64% and 4% to initiating event probability, respectively.
- Total initiating event probability for CRDM ejection in the preceding year is 2.7×10^{-2}.

Conditional Core Damage Probability given initiating event - SPAR Model and licensee provided values are about 3×10^{-3} for MLOCA

NRR-estimated ΔCDF - 8×10^{-5}

Issues/uncertainties - Crack propagation rates of wetted nozzles. Detectable leaks (>1 gpm) before rupture. Time that nozzle was initially wetted. Some in NRR want to remove this from SDP analysis because circumferential crack had not progressed to near rupture.

ENLARGED CAVITY RUPTURE - LLOCA

NRR Initiating event frequency - calculated from the probability that DB could have operated longer, thus had more time for cavity growth

NRR Analysis:

- Created 7x7 matrix with axes
  - availability factors for the 7 B&W plants
  - plant age of the 7 B&W plants
- Calculated the 49 possible ages (EFPYs) for Davis-Besse
- Built probablity distribution using 49 ages
- Bounding Corrosion Rates are 2"/yr (3.5 to 7.5 years to LLOCA) and 7"/yr. (1 to 2 years to LLOCA)

NRR initiating event frequency range - **<0.02 to 0.7 / yr.**

ENLARGED CAVITY RUPTURE - LLOCA (Continued)

Conditional Core Damage Probability given initiating event - SPAR Model estimate is 2×10^{-2}, licensee provided value is about 1×10^{-2} for LLOCA

NRR-estimated ΔCDF - $<2 \times 10^{-4}$ to 9.5×10^{-3}

Issues/Uncertainties

- Hypothetical plant operating profile. Detection probabilities. Leak before break.

- Bounding corrosion rates

  2"/yr - 4.5 to 7.5 years until rupture
  7"/yr. - 1 to 2 years until rupture

- Question - Was there an increased likelihood (above nominal) of a LLOCA?

- OERAB has told NRR that we <u>do not agree</u> with the approach taken.

Lessons Learned Report Completed in October

Observations and Conclusions

- NRC recognized potential for event 10 years ago
- Event was preventable
- Visual inspections are incapable of characterizing nozzle cracking
- Enhanced leakage detection needed
- Presence of boric acid requires action
- Multiple Davis-Besse performance issues indicate inattention to safety issues

Based on Predecisional Draft SDP (October 2002)

Lessons Learned Report Completed in October

Recommendations provided in 10 areas:

(1) inspection guidance;
(2) NRC and industry processes to assess operating experience;
(3) industry code inspection requirements for RCPB components (ASME requirements);
(4) assessment of NRC programs, processes, and capabilities;
(5) NRC staff training and experience;
(6) technical specification requirements related to RCPB integrity;
(7) reactor coolant system leakage monitoring practices and capabilities;
(8) stress corrosion cracking and boric acid corrosion technical information and guidance;
(9) NRC licensing process guidance development and implementation; and
(10) previous NRC lessons-learned reviews.

Lessons Learned Report:
NRC Operating Experience Assessment Recommendations

3.1.6 (1) The NRC should take the following steps to address the effectiveness of its programs involving the review of operating experience:

(1) evaluate the agency's capability to retain operating experience information and to perform longer-term operating experience reviews;

(2) evaluate thresholds, criteria, and guidance for initiating generic communications;

(3) evaluate opportunities for additional effectiveness and efficiency gains stemming from changes in organizational alignments (e.g., a centralized NRC operational experience "clearing house");

(4) evaluate the effectiveness of the Generic Issues Program; and

(5) evaluate the effectiveness of the internal dissemination of operating experience to end users.

Lessons Learned Report:
NRC Operating Experience Assessment Recommendations
(Continued)

- 3.1.6 (2) NRC should update its operating experience guidance documents
- 3.1.6 (3) NRC should enhance the effectiveness of its processes for the collection, review, assessment, storage, retrieval, and dissemination of foreign operating experience (3.1.6)
- 3.1.1(1) The NRC should assemble foreign and domestic information concerning Alloy 600 (and other nickel - based alloys) nozzle cracking and boric acid corrosion from technical studies, previous related generic communications, industry guidance, and operational events. Following an analysis of nickel based alloy nozzle susceptibility to stress corrosion cracking (SCC), including other susceptible components, and boric acid corrosion of carbon steel, the NRC should propose a course of action and an implementation schedule to address the results.

Davis-Besse ASP history

- Much higher than average number and risk of events from 1977 to 1988s (16 events)
- No ASP analyses between 1988 to 1998
- Two ASP analyses in 1998
  - Manual Trip due to CCW leak and de-energizing buses (CCDP=1.4E-5)
  - Tornado-induced LOOP (CCDP=5.6E-4)
- Only current ASP event is Vessel Head Degradation

Recent Davis-Besse ASP Analysis

Tornado-induced LOOP (CCDP=5.6E-4)

- Tornado touchdown causing a LOOP (06/24/98)
- A few minor, easily-recoverable equipment failures occurred
- No significant human errors were cited

Manual Trip due to CCW leak and de-energizing buses (CCDP=1.4E-5)

- Maintenance error causes loss of essential bus D1, nonessential bus D2, and the SBO DG (10/14/98)
- Loss of CCW pump 1-2 and subsequent time delay of CCW pump 1-1 caused the CCW rupture disk to leak
- Operators tripped the reactor due to loss of CCW inventory
- Operators isolated the leak and restored CCW to the RCPs and CRDMs

DAVIS-BESSE ASP ANALYSIS

LER Number	Event Date	Event	CCDP	Event Type
346/98-011	10/14/98	Manual reactor trip due to CCW leak and de-energizing Buses	1.40e-5	Trip
346/98-006	06/24/98	Tornado cause reactor trip and LOOP	5.6e-4	LOOP
346/88-007	03/04/88	CCW valves drift closed	1.6e-6	Trip
346/87-015	12/07/87	Trip due to loss of instrument air w/ TBS valves open	7.6e-7 1.0e-4*	Trip
346/87-011	09/06/87	Trip with 13.8 kV bus unavailable	5.8e-6 6.1e-4*	Trip
346/85-013	06/09/85	LOFW and AFW fails	1.1e-2 3.1e-2*	Trip
346/85-002	01/15/85	LOFW and AFW train	3.0e-4 8.3e-4*	Trip
346/84-013	09/11/84	Trip and AVV fails	4.1e-6 7.2e-5*	Trip
346/84-003	03/28/84	SFRCS transient	1.5e-4 4.1e-4*	Trip
346/83-038 346/83-040	07/25/83	Trip with AFW pump inoperable and failure of two SFRCS channels	8.2e-5	Trip
158860	04/19/80	Loss of two essential buses	1.4e-3	UNIQ
171667	02/26/80	Loss of vital bus	1.7e-3	LOOP
146744	01/12/79	Loss of vital bus while at power	9.9e-4 <2.2e-4**	LOFW
133706	12/11/77	AFW pumps inoperable during a test	2.5e-2 <2.7e-4**	LOFW
132927	11/29/77	LOOP	1.3e-3 <3.9e-4**	LOOP
130788	09/24/77	Stuck open PORV	1.3e-3 <1.0e-2**	LOCA

* Sum Probability (Includes Core Vulnerability Sequences)
** ORNL Revised Value

SUMMARY OF SDP RESULTS - VERY PRELIMINARY

NRR Risk Estimates

Rupture Type	*LOCA Size*	*Initiating Event Frequency*	*CCDP*	*ΔCDF*
as-found cavity	medium	$<2\times10^{-4}$	2.91×10^{-3}	$<6\times10^{-7}$
enlarged cavity	large	0.02 to 0.88	1.08×10^{-2}	2×10^{-4} to 9.5×10^{-3}
nozzle ejection	medium	2.7×10^{-2}	2.91×10^{-3}	8×10^{-5}