PARSONS

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November 10, 1997
Docket No. 50-336
Parsons NUM2-PPNR-0768-L

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Millstone Nuclear Power Station Unit No. 2
Independent Corrective Action Verification Program (ICAVP)

Gentlemen:

This letter transmits summaries of telephone conferences between Parsons Power Group Inc., the U. S. Nuclear Regulatory Commission, NNECo and NEAC on October 23, October 28, October 30 and November 4, 1997. The purpose of these telephone conferences were as follows: October 23 - Containment Spray Calc., Appli. of Safety Guide 26/Reg. Guide 1.26, Use & Control of SF Forms, Spec. Revision, Jumper Device Index No. 2-96-041, Teledyne HPSI Class 1 Piping Stress Analyses, Thermal Margin/Low Power and Power Density Trips; October 28 - Penetrant Examination, HPSI Pump Testing SP2604A&B Procedures, Steam Generator Replacement Project, Seismic Evaluation Report (SQUG), OP2344A- 480 Volt Load Center, Licensing Commitments; October 30 - Corrective Actions, Containment Sump Level, Steam Generator Replacement Project, Exclusion of Faulty/defective Parts, Corrective Action Closeout; and November 4 -PDCR 2-121-81, Reg Guide 1.97 Requirement on Nuclear Flux Measurements, Accumulator Tank Support Steel, ASME Section XI List of Repair/replacements, Tier-2 Accident Mitigation System Review RAI's.

Please call me at (610) 855-2366 if you have any questions.

Sincerely,

Daniel L. Curry

Daniel L. Curry
Parsons ICAVP Project Director

DLC:djv

Attachments
1. Telephone Conference Notes from October 23, 1997
2. Telephone Conference Notes from October 28, 1997
3. Telephone Conference Notes from October 30, 1997
4. Telephone Conference Notes from November 4, 1997

A001/1

cc:
E. Imbro (2) - USNRC
H. Eichenholz - USNRC
R. Laudenat - NNECo
J. Fougere - NNECo
Rep. Terry Concannon - NEAC
Project Files

ADMINISTRATIVE CONFERENCE NOTES
October 23, 1997

PURPOSE: **Administrative telephone conference with NNECo, NRC , NEAC and Parsons to discuss:**

- Containment Spray Calculation
- Applicability of Safety Guide 26 / Regulatory Guide 1.26
- Use and Control of SF Forms
- Specification Revision
- Jumper Device Index No. 2-96-041
- Teledyne HPSI Class 1 Piping Stress Analyses
- Thermal Margin/Low Power and Power Density Trips
- NI Calibration at Startup

Date: **October, 23 1997**
List of Attendees:

NNECo		**NRC**	**NEAC**	**Parsons**
Joe Fougere	Manager, ICAVP	Steve Reynolds		Wayne Dobson
Fred Mattioli	Supervisor, MP2 ICAVP			Eric Blocher
Bob Borchert	Supervisor - Technical Support Eng			Mike Akins
Rich Ewing	Supervisor - Design Engineering			John Strange
Willie Williams	Engineer - CMP			Jaun Cajigas
Lou Chiarizia	Engineer - MP2 Projects			Bob Moyer
Marty Van Haltern	Supervisor - Thermal Hydraulics			Richard Moyer
Dave Bajumpaa	Engineer - Thermal Hydraulics			Samir Serhan
Mike Gancarz	Engineer - ABB/CE			Paul Schmitzer
Robert McBeth	Engineer - CMP			
Chi Wu	Engineer - MP2 Fuels			

1. Containment Spray

BACKGROUND: ABB Calculation no. 006-AS95-C-017 documents the containment LOCA and MSLB analyses of record. The MSLB analysis uses the SGN III built-in containment spray efficiency per a CCT(32) input of zero. The LOCA analysis spray data from the LOCA CONTRANS input is used in a similar fashion. We received SGN-III Figure 6 which apparently represents the code's built-in spray efficiency data.

QUESTION: How is the SGN-III and CONTRANS analysis spray efficiency data related to the MP-2 containment spray nozzle design and design flow rate? The design flow rate for the system is 1300 gpm per pump and 700 microns drop diameter at this flow rate.

Discussion

- Topical Report CENPD-140-A explains that the spray efficiency is based on a 1000 micron drop diameter, and is supported by actual test data. Millstone's 700 micron drop diameter would be conservative.
- Millstone confirmed that they are currently redoing the LOCA and Main steam line break analysis. These should be complete in December.

2. Safety Guide 26 / Regulatory Guide 1.26 (from 10/16 conference)

BACKGROUND: It is not clear to what extent Safety Guide 26 / Regulatory Guide 1.26, "Quality Group Classification and Standards" is applicable to Millstone Unit 2.

LB# 3350 From PI-07 for AFW gives two different statements of compliance with Safety Guide 26. The first reference is to Section 5.2 of the SER dated May 10, 1974, from which it is concluded that Safety Guide 26 / Regulatory Guide 1.26 is applicable to Millstone Point Unit 2. The second is referenced to a response to question 4.7 of NU Amendment No. 15 to License Application dated February 16, 1973 from which it is concluded that only certain portions of the AFW System are in compliance with SG 26.

Although these items relate specifically to the design codes identified in SG 26 for the different quality groups, it is not clear to what extent the guide as a whole is applicable to Millstone Unit 2.

In a letter to the Commission dated September 1, 1972, "Request for Full-Term Operating License", it is stated that 'NUSCO Quality Assurance Manual implements Safety Guide 26 as follows:

1. Part I, Section 2A of the manual provides guidelines to the engineer (designer) for determination of levels in accordance with the Safety Guide.
2. Appendix III of the manual delineates the various levels as determined in Part I, Section 2A.

QUESTION: What does the NUSCO Quality Assurance Manual cover in relation to the guide? How are the Quality Group classifications indicated for the systems and interface of components, and what are the requirements of the interface between systems of different classifications? Is this performed to any industry standards?

Discussion

- Millstone is not committed to Reg. Guide 1.26 except as stated in its position paper. FSAR Table 4.2-4 lists the quality classes.
- When looking at P&ID, one can tell the piping class by pipe identification number. Specification ME-688, (sometimes referred to as the MS3 spec) identifies how to read and interpret the pipe number. For Millstone, the design code class = the safety class. The P&ID does not contain any class boundary information.
- By looking in MEPL, one can determine equipment classification. Specification ME-944 addresses MEPL.
- The discussion regarding Safety Guide 26 in the 9/1/72 "Request for Full-Term Operating License" letter, is only applicable to Millstone Unit 1, not Unit 2.
- The NUSCO Quality Assurance Manual makes no commitment regarding the safety guide.

3. SF Forms (from 10/16 conference)

BACKGROUND: We have seen the following forms used to prepare PCDR's and PDCE's:

SF 305. All revisions. "PDCR Checklist". Procedure and Form.

SF 325. All revisions. " Notice of Design Change". Procedure and Form.

SF 327 All revisions. "Plant Design Change Request". Procedure and Form.

SF 359 All revisions. "PDCR Evaluation". Procedure and Form.

QUESTION: Were these SF forms part of an approved design control program? Was the use of these forms governed by a procedure? If so what were the procedures? Are these forms or similar forms currently in use? Please provide a history as to the use of these forms.

RESPONSE: SF forms were part of NGP 3.03 and various Admin. Control Procedures, (ACP). The DCM manual uses similar forms. NNECo will provide a document listing the forms and the controlling procedures.

4. Specification Revision

In reviewing GRITS for selection of a specification sample, we have run across a number of examples where the specifications are in some level of revision and there are no DCN's posted against the specification.

Examples are:
Specification 25203-SP-EE-101 - Specification for Furnishing Construction Services to Millstone 2 - Revision 4;
Specification 25203-SP-EE-105 - Specification for Spare Reactor Coolant Pump Motor Millstone Unit 2 - Revision 7
Specification 25203-SP-EE-095 - Specification for Electrical Installation Requirements for the Shutdown Cooling Isolation Valve

In light of paragraph 3, of NGP 5.11, how can a specification be revised without a DCN?

RESPONSE: NGP 5.11 was canceled in 6/95. Chapter 6 of the Design Control Manual, (DCM) allows a specification to be revised without using a DCN.

5. Jumper Device Index No. 2-96-041

BACKGROUND: Jumper Device Index No. 2-96-041 consists of two jumpers installed to allow the Enclosure Building Filtration Heaters to operate with the Containment Hydrogen Purge valves open or close. (Normally any one of the four Purge valves opening will turn off both heaters. This is to prevent an explosion when exhausting hydrogen.)

The Technical Evaluation for the Bypass Jumper states "A common mode failure has been found with the connection between the four Hydrogen Purge valves and the Enclosure Building Filtration Heaters. To solve the common mode failure *for the short term*, (emphasis added) during modes 3,4,5 and 6, the Enclosure Building Filtration Heater will be on whenever fan F25A or F25B are in service.

QUESTION:

What is the common mode failure that the Jumper Bypass is solving during the modes stated above?

The Jumper Devices were installed in April, 1996. What action has been taken to provide a permanent solution?

Discussion

The jumper devise was approved in response to a LER written when it was discovered that if a purge valve failed to close, and then a LOCA occurred, the Enclosure Building Filtration Heaters would not work. The heaters are needed post LOCA to protect the charcoal filters. A permanent modification has been approved to correct this situation, M2 97 013. Procedure OP 2313C was affected by this change.

6. Teledyne HPSI Class 1 Piping Stress Analyses

Regarding the Teledyne HPSI Class 1 piping stress analyses and Stress Reports that we have received to date, most of the calculations are of 1975 vintage. No Calculation Change Notices (CCN's) have been supplied with these calculations. However, in at least one case (reference RAI-0665), there is an indication that other analyses exist for this piping.

Based on the above, are the supplied Teledyne calculations the latest calculations for the piping system, or do others exist that may either augment or supersede these calculations?

Discussion

- NNECo is sending Parsons any CCN written for calculations we request. However, CCN use started in the late 1980/early 1990 time frame.
- For the RAI referenced, NNECo reviewed the PDCR and found that a memo was used to justify the removal of the support. NNECo is investigating if a CR should be written.
- Prior to the CCN, nothing tracked a calculation change. A memo could have been used to justify a change, and it may NOT have been reflected in the calculation of record.
- In response to a related question, NNECo will look up the PA 81-005 file to find information on locked up snubbers. Parsons will prepare an RAI to request this information.

7. Thermal Margin/Low Power and Power Density Trips

Many accident analysis use a program to calculate the Thermal Margin/Low Power and Power Density Trip points. These values are then used in the analysis to consider the timing of the accident and its severity. The formulas used for these trip point calculations are not referenced in the analysis. For the accident analysis to be valid, the formula used must be the same as used in the plant calculators. Is there documentation available which can be used to verify that the same formulas were used?

Discussion

- To clarify the above question, The accident analysis calculations determine the conditions at which a plant trip should occur. How does Millstone verify that the plant electronics will trip the plant at the same conditions
- Thermal Margin/Low Power and Power Density Trip point logic is hard wired. I&C procedures setup and validate that the plant will trip as required by the accident analysis.

8. NI Calibration at Startup

Procedure EN21022 paragraph 1.7.4-b requires verification of excore response when between 2% and 5% power but does not state how this is done. NU's response to SOER 90-3 stated that Reactor Engineering uses core differential temperature to verify the NI calibration on startup. How is the verification required by EN21022 performed and what controlling document insures the methodology includes this core differential temperature check?

ADMINISTRATIVE CONFERENCE NOTES
October 23, 1997

Discussion

- The step in procedure EN21022 paragraph 1.7.4-b is a result of a 1984 incident where leads on an excore detector were reversed. Verification of the response simply ensures that the excore detector is working properly.

- Initially, at the start of a new fuel cycle, the NI is setup based on evaluation of excore current readings versus flux levels during the previous cycle. Weighting factors adjust for the new cycle based on predicted fuel flux density from the reactor core analysis. At 20% power a calometeric measurement is performed to check plant power level and the NI is adjusted as needed. This is repeated at 30% power level.

- Core ΔT is an input to the reactor protection system. At low power levels ΔT is small so it is not a very accurate measure of power.

ADMINISTRATIVE CONFERENCE NOTES
October 28, 1997

PURPOSE: **Administrative telephone conference with NNECo, NRC , NEAC and Parsons to discuss:**

- **Penetrant Examination**
- **HPSI Pump Testing SP 2604A & B Procedures**
- **Steam Generator Replacement Project**
- **Seismic Evaluation Report (SQUG)**
- **OP2344A, 480 Volt Load Centers**
- **Millstone Unit 2 Licensing Commitments**

Date: **October, 28 1997**

List of Attendees:

NNECo		**NRC**	**NEAC**	**Parsons**
Joe Fougere	Manager, ICAVP	Steve Reynolds		Wayne Dobson
Fred Mattioli	Supervisor, MP2 ICAVP			Eric Blocher
Steve Wainio	Supervisor - Design Engineering			Greg Cranston
Bill Price	Supervisor - Design Engineering			Cliff Marks
Willie Williams	Engineer - CMP			Don Marks
Sal Orefice	Supervisor			Ray Thomas
Mike McDonald	Engineer - CMP			Ken Gabel
John Bemis	Engineer - MP2 Technical Support			Samir Serhan
Rick Bonner	Supervisor - MP2 Operations			Larry Collier
Prem Godha	Engr. - Nuc Materials Engineering			
Scott Duplantis	Engr - Nuc Materials Engineering			
Charles Peterson	Engr - Nuc Materials Engineering			
Irving Tsang	Engineer - CMP			

1. Penetrant Examination

Has a high temperature (> 125 deg. F) and/or a low temperature (< 60 deg. F) qualification test been performed for the liquid penetrant procedures NU-LP-1 and NU-LP-3 as required by ASME Code Section V Article 6 Subarticle T-680, or has relief been granted from this requirement either through an approved code case or some other regulatory document?

RESPONSE: The procedures were qualified to the low temperature requirement by following a practice used by other companies, i.e. the metal is cooled to less than 60 deg. F. When the liquid penetrant is added, it has such a thin film that its temperature immediately drops and matches the metal.

2. HPSI Pump Testing SP 2604A & B Procedures

- Are the flow rate, pressure and vibration measuring instruments analog or digital in construction detection and indication?
- What is the maximum range of each of these instruments?
- What are the incremental changes for each of the instruments at their test measurement points?
- Is a Post Test calibration routinely performed on each of these instruments or is the calibration performed on a specified periodic basis?
- If instruments are found to be out of calibration, what is the course of action?

Discussion

- Min. flow instruments are digital reading 0.1gal. and calibrated ± 2%. High flow is analog with a digital readout, 0-300gpm range, ± 2% full scale. Vibration instruments are analog with digital readout, 0-30 Khz range, ± 5%.

- Flow and pressure instruments are calibrated every 18 months. Vibration instruments are calibrated annually.

- If an instrument is found to be out of calibration, evaluations are done to determine: 1) the validity of the measurements taken and 2) the operability of the equipment. Appropriate actions are taken based on the results of the evaluations.

3. Steam Generator Replacement Project

a) How was the impact on the feed water system supply lines evaluated and documented with respect to increased vessel mass, new steam generator CG, and changes in thermal as well as seismic design inputs? Reference: piping line number 18" EBB-6; pipe stress analyses problem numbers 21 and 22.

b) How was the impact on the instrument tubing and tubing supports including location (orientation and elevation), rerouting required, fitting changeouts evaluated and documented? How were any performed modifications documented (i.e., DCN's) and justified from a calculation standpoint. What document(s) identifies actual elevations for all "Instrument Nozzles" and what documentation (i.e., DCN's) was used to justify these locations for a setpoint calculation(s)?

Discussion

- The NSSS was evaluated including new CG, thermal characteristics, and motions versus elevations. This information was contained in a single transmittal to NNECo which was provided to Fluor Daniel for revision to the feedwater analysis and for the tubing work.
- If Parsons identifies a tubing DCN of interest, NNECo will track down the calculation that supports the DCN.

c) How were changes on thermal movements for SG nozzles evaluated, and where can these design inputs be found?

e) What was the drawing control process? Specifically what is the relationship of the Fluor Daniel drawings to the NNECo drawings (i.e., 86242-28408-1022 vs. 25203-28408-1022).

Discussion

- For this large project, the original drawings were copied, the normal 25203 id number was replaced with the PA#, (86242) and these were provided to the A/E for their use. After the job was complete and ready for as-building, the PA numbered drawings were reconciled with the record drawings and incorporated into a revision of the 25203 series drawings. During the job, the PA# drawing revisions were controlled via rev. a, b,c, etc. All of the PA# DCNs have been incorporated into the record drawings.
- Currently Raytheon is doing a drawing update backlog reduction effort. For Unit 2 there are approx. 2500 drawings with outstanding DCNs, (none of which are Ops Critical). Of these, approx. 700 need to be need to be complete prior to restart.
- Prior to 1992, NNECo could change a drawing with out a DCN. They would change the drawing directly with a revision instead of issuing a change notice against a drawing, which would later be incorporated into a drawing revision.

d) We need clarification on where is the cut point line location on drawing 25203-29145 sht. 296 for the steam generator replacement component (lower assembly).

f) What is the relationship between Fluor Daniel specifications NU 7-K012/0:830100 70200/0 (instrumentation) and NU 50-5002/0: 830100 50955/0 (root valves) and the respective Bechtel Specs 7604-MS-66 and 7604-MS-64? Which one is the design basis for the steam generator replacement? <u>RESPONSE</u>: NNECo will provide info during the 10/30/97 Conference.

4. Seismic Evaluation Report (SQUG)

The Seismic Evaluation Report (SQUG) walkdown identified outliers for equipment qualification. The response to these outliers was recommended hardware changes listed in the Seismic Evaluation Report. What change control documents ensure that these hardware changes are implemented?

Discussion

- Closure of 87-02 will be done via a letter to the NRC when everything is complete. The actual modifications are accomplished via the Design Control Manual.
- Relocation of a relay is still pending. There was one relay that NNECo thought was a problem, but it was later determined that the relay was class 1E and qualified.
- NNECo can cross reference the 87-02 closure to specific modifications

5. OP2344A, 480 Volt Load Centers

Regarding OP2344A, 480 Volt Load Centers, Section 5.17 - 5.29

- How often are the mentioned jumpers used?
- Does the Temporary Power to the RWST Heat Trace maintain operability?

RESPONSE: The jumpers are used once per refueling outage to de-energize a load center for maint. while still providing power to items which need power to support the outage. The jumpers are provided for in the procedure because they are used regularly. The RWST Heat Trace operability depends on temperature, not electrical power.

6. Millstone Unit 2 Licensing Commitments

In several different documents we have noticed terminology which sounds like there is a data base for Millstone licensing commitments. For example, UIR 2740 contains statements like, "Commitment record XXXX" states ..... Does a commitment database exist? Is this terminology referring to something more than LIST or the PI-06 evaluations?

RESPONSE: Yes, a commitment database exist. It is a ACCESS database created for the CMP. PI-06 populated this database. Parsons has a copy of the database. Its file name is CMPMASTER.DNB.

ADMINISTRATIVE CONFERENCE NOTES
October 30, 1997

PURPOSE: **Administrative telephone conference with NNECo, NRC , NEAC and Parsons to discuss:**

- **Corrective Actions**
- **Containment Samp Level**
- **Steam Generator Replacement Project** (follow-up from 10/28)
- **Exclusion of Faulty/defective Parts**
- **Corrective Action Closeout**

List of Attendees:

NNECo		**NRC**	**NEAC**	**Parsons**
Joe Fougere	Manager, ICAVP	Steve Reynolds		Mike Akins
Fred Mattioli	Supervisor, MP2 ICAVP			Eric Blocher
Bill Price	Supervisor - Design Engineering			Greg Cranston
Willie Williams	Engineer - CMP			Samir Serhan
John Bemis	Engineer - MP2 Technical Support			Ken Gabel
John Resetar	Manager - MP2 Corrective Actions			Ray Thomas
Mark Suprenant	Manager - Procure Engineering			Dan Wooddell
Lou Chiarizia	Engineer - MP2 Projects			Wayne Choromanski
Phil Higgins	Engineer - MP2 Design Engineering			Mark Fitzgerald
Gary Komosky	Engineer - MP2 ICAVP			

1. Corrective Actions

For UIRs that were closed as being picked up by PI-07 review by NNECo, we can not find any reference in PI-07 where numerous UIRs were addressed. Specifically UIR's 1052,1072,1104,1286,1398, and 1616. We expect that other UIR's fall into this same category. How did NNECo documented closure of UIRs as part of PI-07?

During preliminary review of the summaries of closed CR's on the selected Tier 1 systems we found CR M2-97-0900 was closed out to a UIR. This CR referenced UIR 2521 as correcting the issue. Does the NNECo process allow closure of ACR's back to UIR's without issuing a specific corrective action under and AR.

Discussion

PI-14 covers the UIR process used by NNECo. Some of the UIRs were written as informational UIRs and were closed because no corrective action was necessary. If corrective action was necessary, then a CR was initiated against the UIR. An AR could have been written that does not attach to a CR. AR can be closed in a CR if the AR is a long term monitoring issue, or if the department head closes it.

2. Containment Sump Level

NNECo indicated that SP-21136 is the procedure which directs personnel to monitor or measure that piping is maintained with borated water filled to an elevation of (-) 24' or higher in order to prevent pressure locking of Containment Sump Isolation Valves , 2-CS-16.1A and 2-CS-16.1B. What documentation would identify the required frequency for this inspection, and that the inspection has actually been performed? We have looked in PMMS for AWOs associated with these valves, but have been unable to locate anything that indicates that this work was performed

ADMINISTRATIVE CONFERENCE NOTES
October 30, 1997

Discussion
SP 21136 has been done on July 21, 25 '96; October 11, '96 and Jan 10 '97. After that time the plant was shut down. The valves are tested per ASME §XI per IWV 34.11 on a quarterly basis. There is a grace period of approximately 22 days (25% on TS) associated with this. This procedure uses a temporary hose that delivers 15-18 gpm to fill the sump. The procedure will be deleted when a modification installs thermal locking hardware. The modification is currently in development and is a startup item.

<u>QUESTION:</u> Concerning LER 95-002-00, Anchor Darling performed an analysis and determined that the maximum pressure lock that containment sump isolation valves 2-CS-16.1A & B could overcome and still open is approximately 150 psi.

- What Containment Side, RWST Side, and Valve Bonnet pressures were assumed for this analysis?
- What were the method and assumptions used to calculate valve thrust?
- What was the value of the reduced voltage factor that was used in the analysis?

Discussion
There is a letter from Anchor Darling pertaining to these valves. It is based on 150 psid for the Bonnet, with 0 psid up- and down-stream of the discs. There are two calculations for the valves. One for the operability of determination for the valves and one for the design conditions of the valves. The operability calculation does not use degraded voltage, while the design calculation does use degraded voltage.

3. Steam Generator Replacement Project (follow-up from 10/28)

d) We need clarification on where is the cut point line location on drawing 25203-29145 sht. 296 for the steam generator replacement component (lower assembly). Specifically we are looking for the distance between the upper and lower tap for the narrow range steam generator level. instrumentation. Also, does DCN M2-5-546-91 address the new of old steam generator?

Discussion
The drawing that contains the dimensional information for the steam generator is 28408 sheet 1022. This is currently in design engineering for incorporation of DCNs. DCN-m2-5-546-91 will be in the current update.

Elevations were developed based on containment elevations measured to the base reference point on the floor. RAI was issued to request the drawing and to receive a response to the question: "What temperatures were the dimensions based on?"

f) What is the relationship between Fluor Daniel specifications NU 7-K012/0:830100 70200/0 (instrumentation) and NU 50-5002/0: 830100 50955/0 (root valves) and the respective Bechtel Specs 7604-MS-66 and 7604-MS-64? Which one is the design basis for the steam generator replacement?

Discussion
Installation was based on a Bechtel Specification 7604-MS-66 and 7604-MS-64 prepared specifically for Millstone 2, and it contains more information than was used. (Note: this was requested under RAI-739)

4. Exclusion of Faulty/defective Parts

Several regulatory documents such as NRC Bulletins, LER's etc. pertain to defective components or subcomponents. In many cases, Northeast was asked to verify whether or not the subject components or subcomponents were installed at Millstone 2, and if so in what capacity, i.e. safety related or non safety related. Most of the documents reviewed were from the mid 80's time frame or earlier.

QUESTION: What barriers does Northeast have in place to ensure the exclusion of these items? Do you have a document which specifically mentions these faulty/defective parts and is it procedurally referenced in the procurement process? Is this issue addressed in the case of commercial grade components? (Note: we will ask for a copy of documentation that addresses this issue)

Discussion

The plant used the NODIL (Nuclear Operations Defective Items List) since the mid-80's for this. The NODIL is updated annually. It was updated in Jan 97. All the fields are "Q" and controlled by a select few. NGP 6.02 points to the use of MPM 3.00 (basic MIMS procedure) and this points to MPM 3.05, the Admin. procedure.

5. Corrective Action Closeout

BACKGROUND:. A review of two closed ACR's does not show traceability or linkage to associated AR's. A closed ACR could be reviewed and found acceptable without knowing if all AR's associated with that ACR were captured and properly closed. For Open ACR's that we will review to determine if we agree that it can be Open for restart, we have not found an electronic way to link associated AR's to ensure all relevant documents have been reviewed to make the correct decision.

For example, ACR 8490 is shown as closed or complete. The AR numbers referenced on the cover are 96002343-01 and 96002343-02. A different document indicates that AR's 96002343-03 and 97001985-17 are also applicable to ACR 8490 and would have to be closed to allow the ACR to go to complete. However, the later two AR's are not referenced by number anywhere in the ACR. Also, none of the AR's are included as part of the closure package sent to Parsons.

A similar situation exists with ACR M2-96-0398 and AR's 96030546-02, 96030546-03 and 96030546-01. The first two AR's are referenced on the cover sheet of the ACR but the third is not referenced at all. None of the AR's are included in the closure package.

The Utility response to RAI-00490--10/8/97 provided a current list of ACR's/CR's and their status, with a summary. However, the associated AR's are not listed.

QUESTION: Is there a Database that captures and ties ACR's/CR's to associated UIR's and AR's so that completion or acceptability for deferral can be determined? How does NNECo track ARs associated with and ACR/CR?

Discussion

In the passport system there is the ability to tie the ACRs and CRs and ARs together. Modes associated with the ACR are identified at the action level. It will be necessary to review all the actions associated with the ACR in order to determine the closure of the ACR and identifying if it must be completed prior to restart. It is possible to sort the actions using the keyword "mid 13".

AITTS has a complete record of corrective actions, note that it does not contain the full document associated with the action, but does reference it.

ADMINISTRATIVE CONFERENCE NOTES
November 4, 1997

PURPOSE: **Administrative telephone conference with NNECo, NRC , NEAC and Parsons to discuss:**

- **PDCR 2-121-81**
- **Reg Guide 1.97 Requirement on Nuclear Flux Measurements.**
- **Accumulator Tank Support Steel**
- **ASME Section XI List of Repair/replacements**
- **Tier-2 Accident Mitigation System Review RAI's**

List of Attendees:

NNECo		**NRC**	**NEAC**	**Parsons**
Joe Fougere	Manager, ICAVP	Steve Reynolds		Wayne Dobson
Fred Mattioli	Supervisor, MP2 ICAVP			Samir Serhan
Steve Wainio	Supervisor - Design Engineering			Rich Glaviano
Bob Borchert	Supervisor - Technical Support Eng			Dan Cardinale
Dave Bajumpaa	Engineer - Thermal Hydraulics			Larry Collier
Jim DiLuca	Design Engineering			Bob Moyer
Paul Wagner	Manager - Design			
Kalvin Anglin	Design Engineering			
Lloyd Baird	Technical Support Eng			
Ken Fox	Supervisor - Design Engineering			
Mike Kai	Supervisor - Safety Analysis			

1. PDCR 2-121-81

PDCR 2-121-81, titled "Redesign of Pipe Support 401106" proposes to modify this support to meet IEB 79-02 and 79-14 criteria. However, within the PDCR, the Safety Evaluation, Design Inputs, work authorization (PA 79-176) etc. appear to address all 79-02/79-14 rework.

- Does this mod cover all 79-02/79-14 rework, all HPSI 79-02/79-14 rework, or just 401106 rework?

RESPONSE: Just the 401106 rework

- If just 401106, is this the only HPSI support that required rework for 79-02/79-14?
- If additional HPSI supports required rework, which ones and which mod/ (mod's) addressed the work?

RESPONSE: 401106 is not the only HPSI support that required rework for 79-02/79-14. In response to RAI 491, scheduled for 11/13/97, other rework for 79-02/79-14 will be provided.

- When could we get the support design calc for 401106, request by RAI-0197, dated 8-5-97?

RESPONSE: 401106 is currently being reanalyzed. It is anticipated that as part of some RWST work, 401106 will be removed.

2. Reg Guide 1.97 Requirement on Nuclear Flux Measurements.

BACKGROUND: Reg Guide 1.97 requires the display of wide range nuclear flux measurements for monitoring attainment of shutdown conditions and the potential for a recriticality accident. The initial design proposed by NNECo (B14034, 3/2/92) utilized all four wide range channels as accident monitoring instrumentation. The current design (SP-MP-EE-0012 12/2/96) takes credit for only two channels (A & D). However, as-built design includes four wide range logarithmic neutron monitoring channels spaced around the core to measure power and rate of change of power in startup and intermediate powe levels. These signals are provided to the operator on the main control boards.

QUESTION:

a) Has an analysis been performed that addresses the ability of the two selected channels to detect approaches to recriticality under conditions which may involve substantial flux tilting (CEA drop or ejection). If yes, please identify and provide the document.

b) Since all four channels are displayed on the main control board, is there a technical reason why only two of the channels are designated as R.G. 1.97 accident monitoring instrumentation?

Discussion

- No analysis has been done that addresses the ability of the two selected channels to detect approaches to recriticality under conditions which may involve substantial flux tilting. For the examples of CEA drop or ejection there would be reactor shutdown with no return to criticality unless there were multiple failures, so such an analysis is not needed.
- A letter to the NRC dated 11/7/95 changed the RG 1.97 designated channels from 4 to 2. Only the A & D channels are EQ qualified due to a problem with the B & C channel cable mineral insulation.

BACKGROUND: Redundancy requirements of R.G. 1.97 are based on the premise that both of the redundant channels are monitoring the same variable, and that the variable will have essentially the same value at both points of measurement. R.G. 1.97 states that "...(beyond redundancy)... It is important that the number of points of measurement be sufficient to adequately indicate the variable value..." This criteria is typically applied to containment temperature, but applies to neutron flux measurements as well.

QUESTION: Has NNECo analyzed the ability of the system to provide correct information (power level and rate of change) to the operator under conditions of single failure of one of the two channels and substantial flux tilting within the core?

Discussion

- No analysis has been done that addresses this area. Normally, the plant will have 4 channels to detect recriticality, except for situations that result in a harsh environment.

3. Accumulator Tank Support Steel

With regard to original equipment, safety related accumulator tanks, such as T123A&B and T124A&B shown on P&ID 25203-26015 Sheet 1(J-7), and T121, T122, T133 & T134 shown on P&ID 25203-26028 Sheet 3(G-4), we are trying to determine what document provides the criteria/instructions for mounting these tanks to their support steel. Spec. 7604-MS-66, Rev.7, "Design Guide For Seismic Class I Instrument Tubing Installation " states in Section 3.7, "If accumulator tanks and check valves are required on control valves, they shall be installed in accordance with this document or Seismic Class I Criteria". A review of MS-66 indicates no instructions for accumulator tank installation. Therefore, what is the Seismic Class I Criteria referred to and is there any other documents that provides the criteria/instructions for mounting these tanks to their support steel?

RESPONSE: The accumulator tanks were purchased to Spec. MS-226. Bechtel calculations 4 & 6 designed the tank supports.

4. ASME Section XI List of Repair/replacements

Reference RAI 426 response, list of ASME Section XI Repair and Replacements. One of the column labels is "Sent to File". What does this indicate? Many of the items completed in the 1992, '93, and '94 time frame have no date listed for "sent to file". The purpose for this question is to determine what documentation is available for the ICAVP review.
RESPONSE: Any item listed with an AWO that has been completed is available for the ICAVP review. "Sent to file" has no relation to the information being sent to nuclear records.

5. Tier-2 Accident Mitigation System Review RAI's

RAI's 423 & 427 were submitted to gather design information on the SI Tanks and the Shutdown Cooling Systems. Additional RAI's will be submitted to obtain design information on the remainder of the Tier-2 Accident Mitigation systems. Based on Parsons review of the information submitted in response to RAI's 423 and 427, we propose the following approach:

1) Parsons will submit specific RAI's for each of the remaining systems. Each RAI will identify a specific parameter (eg, flow) and request parameter-specific information that can be determined from sources available to Parsons. This information should be provided to Parsons within 2 weeks of request.

2) In addition, we request that the NNECo System Engineer review the RAI and:
  a. identify additional parameter-specific information relevant to the system,
  b. identify calculations listed on the RAI which have been superseded or that are being revised.

3) Is there a limit to the number of documents that can be requested on an RAI? If we need 50 calcs, should we make that into multiple RAI's?

Discussion

The above approach is acceptable. One RAI is desirable, there is no limit to the number of documents that can be requested on a RAI.

4) Several FSAR Chapter 14 analyses are being updated by NNECo. MSLB is an example. Parsons is performing the Tier-2 review using the current analyses. We will review the revised analyses near the end of the Tier-2 review and address any changes that have been incorporated into the updated analyses. We would like to obtain the updated analyses as soon as possible (by early December, if possible). Which analyses are being updated, and can we get copies of the updated analyses by early December?

Discussion

- NNECo is currently preparing inputs for the Chapter 14 analyses and reviewing the schedule. It is unlikely that the updated analyses will be done by early December.
- The following are being reanalyzed:
  - Chapter 14.1 - Excess Load, Main Steam Line Break
  - Chapter 14.2 - Loss of External Load, MSIV Closure
  - Chapter 14.6 - Small LOCA, Large LOCA
  - Chapter 14.8 - Main line Break/Containment Analysis
  - Startup Rod Withdrawal Accident
- NNECo will send Parsons the schedule when it is available
- NNECo will send Parsons the revised analysis inputs when they are available. Parsons can start their review with just the inputs before the re-analysis is completed.