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U.S. NUCLEAR REGULATORY COMMISSION

June 30, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
FSAR Review Meeting
NRC Docket Nos. 50-454/455/456/457

Dear Mr. Denton:

Enclosed are minutes of the meeting held on May 11, 12, and 13, 1981, in Bethesda to review portions of the draft Byron/Braidwood SER prepared by the Mechanical Engineering Branch. Resolution of open items from the draft SER involved discussion with the staff, presentation of technical information, and/or agreement to provide FSAR revisions. A summary of the discussion, presentations, and resolution of each open item is included as Attachment A. In addition, a list of FSAR revisions and/or commitments to provide additional information is included as Attachment B. A copy of the meeting agenda defining the subject matter of the open items is included as Attachment C.

As a result of this meeting, all but nine of the open items in the SER draft have been resolved. The nine open items are described below:

N4 - Westinghouse is to provide a comparison to the NRC for their review.

B1 - This item involves seismic re-evaluation of the station, which has not yet been completed.

N17 - This item is a disagreement between Westinghouse and the NRC.

N15, B16, B18, N23, N25 - All of these items are related to asymetric loading. The analysis is scheduled for completion in fall, 1981.

B001
S/1

C1 - The programs for preservice testing of pumps and valves and for inservice testing of pumps and valves will be submitted to the NRC in early 1982 and September 1982, respectively.

Please address corrections to these minutes and further questions regarding this matter to my office.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator
Pressurized Water Reactors

Enclosure

2236N

Attachment A

Minutes of NRC MEB SER Meeting

Byron/Braidwood

May 11-13, 1981

Item Nl (SER Page 1, Section 3.6.2, Paragraph 3)

A comparison of the Byron/Braidwood design transients and the WCAP-8082 transients was presented. It was pointed out that the Byron/Braidwood transients include all of the WCAP-8082 transients plus a number of additional transients.

Use of the reference analysis was also explained, indicating that the reference analysis utilized the same methods and criteria as WCAP-8082 except that the Byron/Braidwood transients were used rather than the WCAP-8082 transients. In this way, the actual plant thermal and OBE moments calculated for Byron/Braidwood have been shown to be less than the moments of the reference analysis. Thus, the number of breaks and their locations determined from the reference analysis are the same as those given in WCAP-8082, and the stresses and the usage factors calculated in the reference analysis are applicable to Byron/Braidwood. It was concluded that the results of the reference analysis are consistant with WCAP-8082, i.e. no additional breaks need be postulated as a result of the Byron/Braidwood transients.

FSAR Section 3.6.5 will be modified to indicate that the reference analysis specifically considers the same transients as Byron/Braidwood and to confirm that the break locations are the same as indicated in WCAP-8082. With these changes and the meeting discussion, this item is resolved.

Item N2 (SER Page 2, Section 3.6.2, Paragraph 4)

It was explained that a detailed fatigue analysis was performed for the loop isolation valve-to-pipe welds using the Byron/Braidwood transients. It was also indicated that the Byron/Braidwood reactor coolant loop (RCL) model included the loop isolation valves in the fatigue analysis to determine the actual plant moments. The fatigue results showed that the Equation (10) stresses were less than 2.4 S_m and that the usage factors were less than 0.01. Therefore, it was not necessary to postulate new breaks in addition to those defined by WCAP-8082.

Based on the above discussion, this item was resolved.

Item N3 (SER Page 2, Section 3.6.2, Paragraph 5)

The responsibilities of S&L and Westinghouse concerning the interface between the design of the primary equipment supports and the RCL stress analysis were discussed. Also, the design flow process between S&L and Westinghouse, assuring that RCL analysis was consistant with the final support design, was explained. It was noted that actual calculated support stiffnesses were included in the RCL model and that the resulting loop loads and displacements were included in the final support design.

Comparisons between the S&L - designed supports for Byron/Braidwood and typical Westinghouse - designed supports were presented showing the similarity between the two designs. It was further indicated that the Byron/Braidwood loop piping, loop layout, and primary equipment are essentially the same as used in the WCAP-8082 and the reference analysis.

FSAR 3.6.5 will be revised to reference FSAR 3.9.3.4.4.1 which discusses the interface between S&L and Westinghouse. This change and the above discussion constitute resolution of this item.

Item N4 (SER Page 2, Section 3.6.2, Paragraph 6)

At the time the Byron/Braidwood loop piping analyses were performed there were no NRC approved computer codes. SATAN IV was in use at that time. Since then the NRC has approved the MULTIFLEX Code for piping and reactor internals analyses. For Byron/Braidwood, the MULTIFLEX Code is used for the internals. For the piping analyses, Westinghouse has done comparisons between MULTIFLEX and SATAN IV and has shown that comparable results are obtained. Westinghouse will provide these comparisons and the SATAN IV modeling scheme to the NRC for their review.

The comparison is scheduled to be completed by the end of July, 1981. Submittal of this comparison will constitute a resolution for this item.

Item B1 (SER page 5, Section 3.7.3, paragraph 6)

A meeting between the applicant and the NRC was held on May 13, 1981 to discuss the adequacy of seismic margins for the Byron/Braidwood plants. Commonwealth Edison will complete its seismic margin reassessment as quickly as possible and submit it as the response to Question 130.06. The concerns expressed by the MEB will be covered in that response.

Item B2/N5 (SER Page 5, Section 3.7.3, Paragraph 7)

Sargent and Lundy indicated that they require the valve vendor to seismically qualify the valves to certain "g" values and that they check the piping analysis to verify that these values are conservative.

Sargent and Lundy was asked if they have reviewed and rejected any seismic analyses for valves. They answered affirmatively and provided an example for NRC review.

Westinghouse discussed the seismic analysis methods it used to qualify equipment and piping for Byron/Braidwood. Equipment with more than one mode below 33 Hz and all piping systems are qualified using the response spectrum analysis technique. All other equipment is qualified using static analysis method. Westinghouse agreed to provide changes to the FSAR that would summarize the discussions of analysis technique.

A question was asked concerning the modeling of valves in piping systems. Westinghouse stated that rigid valves are modeled as a mass on an extended structure. Non-rigid valves are modeled as flexible in the piping analysis, Westinghouse requirements for rigid valves and its review of valve vendor reports were discussed. Westinghouse will provide a statement of its valve modeling techniques in the FSAR.

Based on these discussions and the changes to be made to the FSAR these items are resolved.

Item B3/N6 (SER Page 5, Section 3.7.3, Paragraph 8)

Sargent and Lundy stated that all seismic restraints in piping systems within Sargent and Lundy's scope of work are considered to be infinitely rigid for analytical purposes.

Westinghouse stated that for all piping systems they are evaluating for Byron, the calculated support stiffness is included in the analysis. Westinghouse will provide a statement to this effect in the FSAR to resolve this issue.

For Byron, Westinghouse is responsible for the analysis of all piping inside containment, and for the safety injection system, residual heat removal system, chemical and volume control system, and containment spray system in the auxiliary building.

Based on this discussion and the FSAR changes which will be made, this item is closed.

Item B4 (SER page 6, Section 3.7.3, paragraph 4)

A discussion of the extent of use of the static load method took place. It became apparent that Section 3.7.3.5 of the FSAR is unclear. We propose to revise it to read:

> No static load method is utilized in the seismic analysis of piping systems.
>
> However, in the seismic analysis of equipment, the equivalent static load method is used if the equipment is not rigid and a dynamic analysis is not performed.
>
> If the fundamental natural period (FNP)...

This will resolve the item.

Item N7 (SER Section 3.7.3, Page 6, Paragraph 9)

Westinghouse stated that all equipment with more than one mode below 33 Hz and all piping systems that they are responsible for (see Item B3/N6) are analyzed with response spectrum techniques. All other equipment is analyzed using static analysis methods. Westinghouse indentified the equipment with more than one mode below 33 Hz as the steam generator, reactor coolant pump, pressurizer, control rod drive mechanisms, reactor internals and fuel. Westinghouse will include this information in the FSAR to resolve this item.

Item N8 (SER Page 6, Section 3.7.3, Paragraph 10)

The method used by Westinghouse for the combination of closely spaced modes has been accepted previously by the NRC (RESAR 41, RESAR 414, numerous plant dockets) as an acceptable alternative to Regulatory Guide 1.92. The NRC Mechanical Engineering Branch will notify the Structural Branch that on this basis this item is considered resolved.

Item B5 (SER page 6, Section 3.7.3, paragraph 11)

Sargent & Lundy described buried cat I structures and the analysis of them, and agreed to provide a write-up to be added to the existing FSAR Section 3.7.3.12 delinating stress limits used for buried piping analysis, as follows:

> "Since all buried essential service water piping falls under subsection NC of ASME B&PV code, section III the following stress limits are met:
>
> | | | |
> |---|---|---|
> | Stresses due to sustained loads | $\leq$ | $1.0S_h$ |
> | Stresses due to occasional loads (OBE) | $\leq$ | $1.2S_h$ |
> | Stresses due to occasional loads (SSE) | $\leq$ | $1.8S_h$ |
> | Stresses due to bending moments caused by soil settlement and/or overburden pressure | $\leq$ | $3.0S_c$ |
>
> For all buried concrete electrical duct runs associated with the essential service water system, the design is in accordance with ACI-318-71 requirement.

Based on the discussion, and the proposed FSAR changes, this item is resolved.

Item N9 (SER Section 3.7.3, Page 6, Paragraph 12)

Westinghouse explained that the damping values in Regulatory Guide 1.61, i.e. two (2) percent for the OBE and four (4) percent for the SSE are utilized. Based on the above explaination and the addition of this information to the FSAR, this item is resolved.

Item B6/N10 (SER Page 7, Section 3.9.1, Paragraph 2)

Sargent and Lundy stated that the PIPSYS (integrated piping analysis system) computer code was bench marked against two public domain computer programs, DYNAL and NASTRAN, and was found to be acceptable. This is documented in FSAR Appendix D.8.

Westinghouse discussed the method utilized in piping analysis for lumping masses and referred to Dr. Lim's paper, "How to Lump the Masses" as the basis for this method. Dr. Lim's paper is referenced in Section 3.7 of the FSAR.

For flexible equipment, Westinghouse utilizes multiple degree of freedom dynamic analysis models (e.g. over 200 degrees of freedom for the steam generator) to assure that a sufficient number of modes are calculated. This method is consistant with SRP 3.7.2.

Westinghouse also provided test results for review that support the validity of their modeling techniques. These results were for piping systems in two plants and compared the analysis results with test data obtained at the plant site.

Based on these discussions this item was resolved.

Item N11 (SER Page 7, Section 3.9.1, Paragraph 3)

No response to this item is required for MEB review.

Item N12 (SER Page 8, Section 3.9.1, Paragraphs 4 and 5)

Westinghouse described the test load method of qualification of the reactor vessel support pad and shoe and the elastic system analysis with inelastic component analysis used for qualification of the reactor coolant pump support foot. Both of these methods are used in lieu of the Appendix F limits of the ASME Code, Section III.

For the test load method, Westinghouse will provided a revised section for the FSAR that will contain more information on the techniques. A revised section for the FSAR will also be provided that justifies the use of the elastic system with inelastic component analysis.

Based on this discussion and the FSAR changes which will be made, this item is resolved.

Item N13 (SER Page 8, Section 3.9.1, Paragraph 6)

Westinghouse uses 4% of critical damping for the SSE seismic analysis fo the reactor coolant system and supports. The justification of this damping value is provided in WCAP 7921-AR, which has been reviewed and approved by the NRC. This WCAP is referenced in the Byron/Braidwood FSAR. Based on this discussion this item is resolved.

Item B7 (SER page 10, Section 3.9.2.1)

The preservice inspection program will include the visual examination of all hydraulic and mechanical snubbers installed on safety related systems. The inspection will verify that the snubbers are installed correctly, and are undamaged.

A list of all snubbers (both hydraulic and mechanical) on safety related systems will be developed. Documentation will be provided to record the inspections conducted on each snubber. The documented inspection will be conducted no longer than 6 months prior to the preservice testing program in order to satisfy preservice testing requirements.

For hydraulic snubbers the fluid will be verified to be at the recommended level and not leaking.

During hot functional testing, snubber thermal movements for systems whose operating temperature exceeds 250°F will be verified. The thermal monitoring program will be included in the test program. The thermal monitoring program consists of visual verification of snubber movements, as indicated on the snubber, from room temperature to maximum operating temperature. If maximum operating temperature is not attained during testing, the amount of movement expected will be calculated by multiplying the movement indicated on the snubber by the ratio of the temperature rise to the test temperature ($\Delta T_m/\Delta T_t$). If snubber movement differs from the expected movement by more than 1/8 inch, as assessment will be made to verify that the snubber will satisfy its design function for the design load.

Based on this discussion and commitment, this item is resolved.

Item B8/N14 (SER Page 11, Section 3.9.2.2)

Review of the dynamic qualification of mechanical equipment will be covered by the Equipment Qualification Branch. No response is required for the MEB review.

Items N15 and N25 (SER Section 3.9. .3, Page 12, Paragraph 2, and SER Section 3.9.5, Page 20, Paragraph 4)

Westinghouse stated that for reactor internals, the LOCA evaluation has always considered the asymmetric loading "inside" the reactor vessel. (See FSAR Section 3.9.7, Reference #7: G. J. Bohm and J. P. LaFaille "Reactor Internals Response Under a Blowdown Accident" First Int'l Conference on Structural Mechanics in Reactor Technology, Berlin, September 20-24, 1971.)

FSAR Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals for Faulted Conditions", describes in detail the reactor internals blowdown analysis and also refers to "Reference #7" noted above. Question 110.62 concerns the asymmetric loads "outside" the reactor vessel. These loads due to cavity pressure along with loop loads were shown to be insignificant with respect or to the reactor internals.

Any additional information required on the internals concerning asymmetric loadings will be submitted with the response to Question 110.62. (Also see the response to item N23.)

The above discussion along with the response to Question 110.62 will resolve this item.

Item B10 (SER page 13, Section 3.9.3.1, paragraph 1)

The loading combinations for ASME Class I component supports were presented from FSAR Table 3.9-2. The analytical method used to combine the loads is by algebraic summation. The signs of OBE and SSE are chosen to maximize the magnitude of the total load.

A summary of the faulted loads which control the design and the resulting largest stress as a percentage of allowable per FSAR Section 3.9.3.4.5 was presented for each component support. In all cases, the actual stresses are less than or equal to the allowable stresses. The staff requested that the highest stressed member for each component support be identified on the FSAR figures with a corresponding qualitative discussion of stress state and nominal margin to failure for each critical member. Commonwealth Edison Company agreed to include this requested information in a future FSAR amendment.

The Staff questioned which allowables were used for bolt materials. Sargent & Lundy stated the faulted allowables were obtained by using factors calculated in accordance with Appendix F of ASME Section III, Article F1370 times the normal allowables. The Staff requested that the allowables be provided in tabular form to ensure they are below 0.90 of yield strength.

The ratio of the faulted allowables to the yield strength was presented for the high strength bolts used in the Class I component supports. The factors calculated in accordance with ASME Section III, Appendix F, result in faulted condition allowables which are less than 90% of yield strength (Sy) in tension and/or shear. Commonwealth Edison Company agreed to include this information in a future FSAR amendment.

The above discussion concerns Class 1 component supports. In addition, tables in FSAR section 3.9 will be revised to reflect the load combination methodology used for Class 2 and 3 piping, equipment, piping supports and equipment supports.

Based on the discussions and submittal of the FSAR revisions, this item is resolved.

Item N16 (SER Section 3.9.3.1, Page 13, Paragraph 1)

The methodology of load combinations and applicable stress limits for Class 1, 2, and 3 equipment was discussed. Several necessary modifications to the FSAR were identified:

- Table 3.9-3 should be clarified indicating what stress limits apply for each of the operating condition classifications.

- The FSAR should include the load combination methodology applicable to the loads identified in Tables 3.9-2 (Class 1) and 3.9-5 (Class 2 and 3).

- The source of the stress limits for Class 2 and 3 equipment should be identified.

- Class 2 and 3 equipment supports stress limits should be provided.

Additional information on load combination for reactor internals is presented in items N18 and N26.

Based on these FSAR changes and the meeting discussion, this item is resolved.

Item B11 (SER page 13, Section 3.9.3.1, paragraph 1)

Sargent & Lundy described their requirements for functional capability for essential piping. All piping systems that are designated essential and are within Sargent & Lundy scope of work, are evaluated using the functional capability criteria outlined in GE's Topical Report #NEDO-21985, September, 1978, which was evaluated and approved for use by the Mechanical Engineering Branch of NRC.

All essential Byron/Braidwood piping will fall within the following range $50 < D_o/t < 100$, or $D_o/t \leq 50$.

For Sargent & Lundy scope piping, this item is closed.

Item N17 (SER Page 13, Section 3.9.3.2, Paragraph 1)

For Class 2 and 3 austenitic steel bends and elbows, Westinghouse and the NRC could not reach a mutually acceptable resolution of stress criteria for functional capability. This item will remain open.

Westinghouse has formed a task force to resolve this issue. The resolution is expected within approximately two months and will be submitted to the NRC for their review.

Item N18 and N26 (SER Section 3.9.3.1, Page 14, Paragraph 2, and SER Section 3.9.5, Page 20, Paragraph 5)

A discussion of reactor internals stresses and deformations was presented. It was indicated that although the Byron/Braidwood internals are not contractually required to meet ASME Code requirements, essentially the design and fabrication requirements of Section NG of the ASME Code have been satisfied. Exceptions to code requirements discussed at the meeting were no code stamp and no plant-specific stress report. Additionally, all stresses and deformation are below allowable limits. Westinghouse agreed to provide a statement in the FSAR indicating the differences between the Westinghouse criteria used for the Byron/Braidwood internals and the ASME Code requirements. Additionally, Westinghouse agreed to provide a statement in the FSAR relative to the acceptability of stresses and deformations for the Byron/Braidwood internals. Although new specific issues have been identified relative to Regulatory Guide 1.20, Westinghouse stated that Byron/Braidwood references Indian Point Unit 2 and Trojan as the prototype plants for internals vibration monitoring. It should also be noted that the Indian Point tests were conducted both with and without fuel assemblies in the core at the time of the vibration monitoring. The vibration levels under actual operating conditions (i.e. with fuel in the core) are typically lower than those observed without the fuel in place.

Westinghouse further stated that when appropriate, e.g. simple beam analysis, LOCA and SSE loads are combined on a reactor internals structural component basis per the SRSS method, and the resultant stress intensities calculated. For more complex structural geometries (e.g. core barrel shell) the stress components due to LOCA and SSE are combined either by absolute sum or by SRSS, preserving the appropriate signs. These stress components are used to determine the stress intensity for the structural component. For the LOCA, the maximum stresses from the time history response are used. Since the seismic stresses are calculated using response spectrum techniques, the responses are unsigned; therefore, when the LOCA and SSE stresses are combined, the most unfavorable sign convention for the SSE is assumed.

Based upon the above, this item will be resolved upon revision of the FSAR.

Item N19 (SER Section 3.9.3.1, Page 14, Paragraph 2)

Regulatory Guide 1.121 "Bases for Plugging Degraded Steam Generator Tubes"

Q1 - NRC requires that a margin of 3 against tube burst as outlined in this R.G. must be maintained.

A1 - Westinghouse uses a margin of 2 against tube failure. The definition of tube failure is plastic deformation of a crack to the extent that the crack opens to a non-parallel elliptical configuration. NRC defines tube failure as tube burst. Since Westinghouse uses a different definition of tube failure we use a smaller margin. The position on R.G. 1.121 will be expanded to include this information.

Q2 - NRC would like the position to reference the tech spec requirement to plug all degraded tubes that have been reduced in wall thickness by 40% of the nominal tube wall thickness.

A2 - The 40% T.S. limit is a reference limit for Westinghouse steam generators. R.G. 1.121 analyses have not been completed for model D4 and D5 steam generators used in Byron/Braidwood. These analyses will be completed prior to first refueling and at that time the T.S. limits will be re-evaluated and this information can then be included in the R.G. 1.121 position, if necessary.

Based on the above, this item is resolved.

Item B12/N20 (SER Section 3.9.3.2, Page 15, Paragraph 2)

Review of pump and valve operability will be covered by the Equipment Qualification Branch. No response is required for the MEB review.

Items N21 and N22 (SER Section 3.9.3.3, Page 15, Paragraphs 2 and 4)

Westinghouse discussed the analysis methods used to evaluate the pressurizer safety and relief valve discharge piping. Additional FSAR information will be provided which describes the hydraulic and structural analysis methods, loading combinations, inclusion of the effects of water slugs from the loop seals, and valve opening sequence. Also to be provided are the makes and types of valves used and their mounting arrangement. With the submittal of this FSAR write-up, this item will be resolved.

Item B14 (SER page 16, Section 3.9.3.3, paragraph 3)

Sargent & Lundy discussed the use of design load factors (DLF) in the design of relief valves. This discussion revealed that the main steam relief valves utilize a DLF less than 2.0. The design basis for utilizing a DLF less than 2.0 was a parametric study based on a dynamic analysis as allowed for in Code Case 1569. A parametric study done for Zion was shown to the NRC to illustrate the basis for utilizing the lower DLF. All other relief valves utilize a DLF of 2.0.

FSAR Appendix A, A1.67 will be revised to reflect this discussion. This discussion and the resulting FSAR change will close this item.

Item B16 (SER page 17, Section 3.9.3.4, paragraph 2)

An assessment of the NSSS component supports for the faulted condition against 67% of critical buckling was presented. The stresses were calculated on the basis of an SRSS combination of LOCA and SSE for the faulted condition. The stresses calculated on this basis are less than 0.67 times the critical buckling stress in all cases except one. The steam generator lower lateral support has one member which is stressed to 0.73 of critical as calculated from ASME Code Appendix XVII interaction equation.

A figure of the steam generator lower lateral support was presented and the member stressed to 0.73 of critical was identified. The stress in this member of the steam generator lower lateral support is primarily due to a jet impingement load resulting in weak axis bending. The ultimate capacity of this wide flange type member which has acceptable width thickness ratios for its flanges would be governed by the plastic capacity of the section and, therefore, the recommendation of the Regulatory Guide 1.124 and 1.130 on critical buckling would not directly apply. The staff agreed with this justification. Commonwealth Edison Company acknowledged that the response to Question 110.50 and Appendix A will be revised to point out this exception to the regulatory guides.

The effects of asymmetric pressurization loads will be assessed in response to FSAR Questions 110.14 and 110.62. This item will remain open pending submittal of the FSAR revisions and responses to Questions 110.14 and 110.62.

Item B17 (SER page 17, Section 3.9.3.4, paragraph 3)

Sargent & Lundy presented a table of operating temperatures for each Class I component support. In linear elastic analysis, the effect of temperature is accounted for by a reduction in yield stress (S_y) and ultimate tensile stress (S_u) as specified in Subsection NF-3229, Appendix XVII Article 1121, and Appendix F Section 1370(a) of the ASME Code, Section III, Division I, Summer 1975 Edition.

The Staff requested the source of these reduced values for yeild stress and ultimate tensile stress. The reduction in yield and ultimate stress are in accordance with ASME Section III, Appendix I or Code Case 1644. Commonwealth Edison Company agreed to include this information in a future FSAR amendment.

Based on the discussion and the FSAR changes to be made, this item is resolved.

Item B18 (SER page 17, Section 3.9.3.4, paragraph 4)

Sargent & Lundy presented a discussion of their analysis of NSSS component supports.

Sargent & Lundy has assessed NSSS component supports for asymmetric pressure loads provided by Westinghouse for breaks in the hot, cold and cross-over legs. We find that these supports are within the limits described in FSAR paragraph 3.9.3.4.5.

We are in the process of completing the assessment for asymmetric pressure loads and will transmit our results in response to FSAR Questions 110.14 and 110.62.

This item will remain open pending completion of the analysis described above, and documentation of the results in the FSAR.

Item N23 (SER Section 3.9.3.4, Page 17, Paragraph 4)

A general discussion describing how asymmetric loads (Question 110.62) are included in the analysis of reactor coolant system piping and components was presented. The FSAR changes necessary to respond to Question 110.62 will be provided. Upon submittal of this information, this item is resolved. (Also see items N15 and N25 for reactor internals.)

Item N24 (SER Section 3.9.4, Page 18, Paragraph 2)

Evaluation of the loading combination and stress limits for pressure boundary components of the CRDMs, and the design criteria and loading combinations used for non-pressure boundary components of the CRDMs were satisfactorily addressed based upon review of the Westinghouse design specification, discussion of testing performed by Westinghouse and Westinghouse licensees, and the contents of the FSAR.

It was agreed that the information provided in the FSAR is sufficient provided a statement is added to the FSAR which (1) states that FSAR Table 3.9.2 is applicable to Class 1 components of the CRDM and (2) discusses the operational testing and experience gained by Westinghouse & Westinghouse Licensees. Upon revision to the FSAR, this item will be resolved.

Item N27 (SER Section 3.9.5, Page 20, Paragraph 6)

Test results were presented by Westinghouse indicating that for strain ranges below 1.8%, the irradiated material has higher fatigue allowable (allowable stress at given number of cycles) than the unirradiated material. The tests performed included both laboratory-irradiated material and irradiated material from operating plants. It was noted that the strain levels in the internals structures are less than 1.8%.

Based upon the information presented, this item was resolved.

Item C1 (SER page 22, Section 3.9.6)

The preservice testing of pumps and valves is performed during system pre-operational testing. The program plan will be submitted with the other portions of the preservice inspection plan. The complete plan is expected to be available in early 1982.

Commonwealth Edison will submit its program for inservice testing of pumps and valves as requested by Question 110.64. This program will include valves between the reactor coolant system boundary and low pressure systems that penetrate the containment.

It is anticipated that the program will be submitted for Staff review in September, 1982.

Item C2

During preoperational testing normal operating modes will be observed for vibration. Engineers familar with the subject piping will visually inspect the lines to determine the acceptability of the steady state vibrations. Abnormal vibration will be noted. If piping system vibration is judged excessive, corrective action will be either:

1. The cause of the excessive vibration will be eliminated.
2. The support system will be modified to reduce the vibration to acceptable limits.
3. The piping will be monitored by instrumentation at locations which appear to be excessive to demonstrate that the measured pipe deflections when converted to stress will not exceed 50% of the material endurance limit selected from the value at 10^6 cycles from the curves of Appendix I-9.0 of Section III of the ASME Code.

Based on the discussion and the commitment stated above, this item is closed.

Attac'hment B

FSAR Revisions Required

Item	Info, Needed	FSAR Section Impacted
N1		3.6.5
N3		3.6.5
N4	SATAN IV vx. MULTIFLEX	
N5		3.7.3
N6		3.7.3
B4		3.7.3
N7		3.7.3
B5		3.7.3
N9		3.7.3
N12		3.9.1
N15, N25	Answer Q110.62	
B10		3.9.3
N16		3.9.3
B11/N17	Open Item on Pipe bends and elbows	
N18, N26		3.9.3, 3.9.5
N19		Appendix A Reg. Guide 1.121 response and analysis needed
N21, N22		3.9.3
B14		Appendix A, Reg. Guide 1.67
B16	Answer Q110.14, Q110.62	3.9.3.4.5, Appendix A
B17		3.9.3.4
B18	Answer Q110.14, Q110.62	
N23	Answer Q110.62	
N24		3.9.4

Attachment C

MEB SER Review Meeting Agenda Items

Monday Afternoon

Item	SER Reference Page	SER Reference Paragraph	Description	Response
N1	1	3.6.2 (3)	Justify break locations from WCAP-8082A	Bob Kelly Brad Maurer
N2	2	3.6.2 (4)	Justify use of WCAP-8082A with loop isolation valves	Bob Kelly Brad Maurer
N3	2	3.6.2 (5)	Describe Westinghouse/BOP interface for component supports	Bob Kelly Brad Maurer
N4	2	3.6.2 (6)	Justify use of SATAN-IV	Bob Kelly Brad Maurer
B1	5	3.7.3 (6)	Demonstrate adequacy of seismic margins (Deconvolution)	J. T. Westermier
B2/N5	5	3.7.3 (7)	Provide natural frequencies for Seismic Cat. I structures	B2-R. J. Netzel N5-Bob Kelly Brad Maurer
B3/N6	5	3.7.3 (8)	Provide discussion of modeling for pipe supports and snubbers	B3-A. A. Deguermendjian N6-Bob Kelly Brad Maurer
B4/N7	6	3.7.3 (9)	Identify equipment for which the natural period is not known	B4-K. L. Adlon N7-Bob Kelly Brad Maurer
N8	6	3.7.3 (10)	Response to Q110.33 part (4) is not satisfactory	Bob Kelly Brad Maurer
B5	6	3.7.3 (11)	Identify buried structures and describe seismic analysis	A. A. Deguermendjian

MEB SER Review Meeting Agenda Items

Tuesday Morning

Item	SER Reference Page	SER Reference Paragraph	Description	Response
B6/N10	7	3.9.1 (2)	Provide basis for selection of number of masses in seismic model of piping and components	B6-A. A. Deguermendjian N10-Bob Kelly Brad Maurer
N11	7	3.9.1 (3)	Specify use of simplified elastic-plastic methods/provide tabular summary of stress ranges	Bob Kelly Brad Maurer
N12	8	3.9.1 (4&5)	Justify use of ASME Section III Appendix F for general components/Specify deformation/displacement limits	Bob Kelly Brad Maurer
N13	8	3.9.1 (6)	Justify 4% critical damping for reactor coolant loop and supports	Bob Kelly Brad Maurer
B7	10	3.9.2.1	Document preservice examination and pre-operational testing program for all snubbers	L. A. Bowen
B8/N14	11	3.9.2.2	Dynamic Qualification of mechanical equipment	B8-K. L. Adlon N14-Bob Kelly Brad Maurer
B11/N17	13	3.9.3.1 (1)	Address functional capability of equipment when service B limits are exceeded (Question 110.40)	B11-A. A. Deguermendjian N17-Bob Kelly Brad Maurer
N19	14	3.9.3.1 (4)	Revise response to Q110.61 regarding tube plugging limit	Mae Wright
B12/N20	15	3.9.3.2	Pump and valve operability assurance	B12-K. L. Adlon N20-Bob Kelly Brad Maurer

MEB SER Review Meeting Agenda Items

Tuesday Afternoon

Item	SER Reference Page	SER Reference Paragraph	Description	Response
N9	6	3.7.3 (12)	Resolve concerns regarding damping valves used for reactor internals seismic analysis	Mark Beaumont
B9/N15	12	3.9.2.3 (2)	Impact of revised seismic input on safe shutdown evaluation	B9-R. J. Netzel N15-Mark Beaumont
B10/N16	13	3.9.3.1 (1)	Provide discussion of load combination methods and max loads, stresses, deformations	B10-R. J. Netzel N16-Mark Beaumont
N18	14	3.9.3.1 (2)	Address reactor internals as requested in Q110.15 and Q110.41	Mark Beaumont
N24	18	3.9.4 (2)	List load combinations and stress limits for pressure boundary items/Specify design criteria and load combinations for non-pressure boundary items	Mark Beaumont
N25	20	3.9.5 (4)	Address asymetric loading on reactor vessel intervals	Mark Beaumont
N26	20	3.9.5 (5)	Clarify manner of compliance with ASME code	Mark Beaumont
N27	20	3.9.5 (6)	Clarify use of unirradiated material properties for fatigue evaluation of reactor intervals	Mark Beaumont

MEB SER Review Meeting Agenda Items

Wednesday Morning

Item	SER Reference Page	SER Reference Paragraph	Description	Response
B13/N21	15	3.9.3.3 (2)	For safety and relief valves discharging to closed system, include effects of water slugs	Bob Kelly Brad Maurer
B14	16	3.9.3.3 (3)	For safety valves that discharge to an open system, justify use of DLF less than 2.0	A. A. Deguermendjian
B15/N22	15	3.9.3.3 (4)	Identify safety and relief valves locations, mounting arrangements, opening sequence, and load combination and stress limits	B15-A. A. Deguermendjian N22-Bob Kelly Brad Maurer
B16	17	3.9.3.4 (2)	Justify use of stress limits 50% greater than normal allowables	R. J. Netzel
B17	17	3.9.3.4 (3)	Identify component supports which are subjected to temperatures greater than ambient	R. J. Netzel
B18/N23	17	3.9.3.4 (4)	Respond to Q110.14 and 110.62: asymetric loads on component supports	B18-R. J. Netzel N23-Bob Kelly Brad Maurer
C1	22	3.9.6 (4)	Submit program for preservice and inservice testing of pumps and valves	L. A. Bowen