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Exelon Nuclear

RS-01-110

January 31, 2001

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

In Reference 1, we submitted proposed changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes would revise the maximum power level specified in each unit's license and the TS definition of rated thermal power.

On September 20, 2000, representatives of Commonwealth Edison (ComEd) Company, now Exelon Generation Company, LLC, and the NRC met to discuss technical issues associated with this license amendment request. In Reference 2, the NRC requested that we formally document the information discussed during this meeting along with some additional information in order to complete its evaluation. In a subsequent teleconference on November 8, 2000, the NRC also requested that additional information be provided regarding the documents reviewed in support of the Power Uprate Environmental Assessment. Our response to these requests for additional information was submitted to the NRC in Reference 3.

In Reference 4, the NRC forwarded a second request for additional information to ComEd which addresses issues in a technical area not addressed by Reference 2. Our response to these requests for additional information was submitted to the NRC in Reference 5.

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In Reference 6, the NRC forwarded a third request for additional information to ComEd which addresses issues in a technical area not previously addressed. In addition, supplemental information to a number of previous NRC questions was also requested. Our response to these requests for additional information is included in Attachment 2. The NRC requested that this response be submitted by January 31, 2001.

Should you have any questions or concerns regarding this information, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Director – Licensing
Mid-West Regional Operating Group

Attachment 1: References

Attachment 2: Response to Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Office of Nuclear Safety – Illinois Department of Nuclear Safety

bcc: NRC Project Manager – NRR – Byron Station
NRC Project Manager – NRR – Braidwood Station
Nicholas Reynolds – Winston and Strawn
Site Vice President – Braidwood Station (w/o att.)
Site Vice President – Byron Station (w/o att.)
Vice President – Licensing and Regulatory Affairs
Director – Licensing
Regulatory Assurance Manager – Braidwood Station
Regulatory Assurance Manager – Byron Station
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Tom Gerlowski – Westinghouse

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
EXELON GENERATION COMPANY, LLC	**)**	**Docket Numbers**
BYRON STATION UNITS 1 AND 2	**)**	**STN 50-454 AND STN 50-455**
BRAIDWOOD STATION UNITS 1 AND 2	**)**	**STN 50-456 AND STN 50-457**

SUBJECT: **Response to a Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director – Licensing

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 30th day of January, 2001.

"OFFICIAL SEAL"
V. LYNN BERNHAGEN
Notary Public, State of Illinois
My Commission Exp. 05/30/2004

V. Lynn Bernhagen
Notary Public

ATTACHMENT 1

References

1. Letter from R. M. Krich (Commonwealth Edison Company) to U.S.NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

2. Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated October 19, 2000

3. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations," dated November 27, 2000

4. Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated November 21, 2000

5. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations," dated December 21, 2000

6. Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated December 22, 2000

ATTACHMENT 2

Response to Request for Additional Information
Regarding a License Amendment Request to Permit Uprated Power Operations
at
Byron and Braidwood Stations

In a letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison (ComEd) Company), "Byron Station and Braidwood Station - Request for Additional Information Regarding the Power Uprate Request," dated December 22, 2000, the NRC requested that additional information be provided (NRC Question Set J). Additional information relating to prior NRC question set responses, requested during follow-up conference calls, is provided following the responses to Question Set J.

Note: References, if applicable, appear at the end of each response and are noted in text with "[ ]".

<u>NRC Question Set J</u>

J.1 *In Section 5.1.1.3 of Attachment E to the reference transmittal, the licensee stated that the very conservative maximum range of stress intensity and cumulative fatigue usage factor (CUF) results for the Byron Station inlet nozzle safety ends were applied to Braidwood Station to simplify the design transient evaluation. Provide the technical basis that the Byron Station results can be applicable to Braidwood Station. Confirm that the American Society of Mechanical Engineers (ASME) Code Section III, 1971 Edition with addenda through the Summer 1973, is the Code of record specified in the UFSAR of both Braidwood Station and Byron Station, and that they were used in the evaluation of the reactor vessel for the power uprate for each plant.*

<u>J.1 Response</u>

The Byron Station Unit 1 maximum range of stress intensity and CUF results are applicable and considered conservative with respect to Byron Station Unit 2 and Braidwood Station Units 1 and 2 based on the following discussion.

1. The inlet nozzles for Byron Station and Braidwood Station have essentially the same design drawing dimensions. The following differences are noted:

   a) The lengths of safe end and weld to the end of the nozzle forging is 3.094" for Byron Station Unit 1 and 3.281" for Byron Station Unit 2 and Braidwood Station Units 1 and 2.

   b) The length of the nozzle thin end to the beginning of the transition taper is 5.625" for Byron Station Unit 1 and 5.438" for Byron Station Unit 2 and Braidwood Station Units 1 and 2.

   c) The overall length of the Byron Station Unit 1 inlet nozzles is 42.5" minimum and the overall length of the Byron Station Unit 2 and Braidwood Station Units 1 and 2 inlet nozzles is 42.313" minimum, for a net difference of 0.187".

d) The < 0.5% difference in the moment arms used in the transfer equations between Byron Station Unit 1 and the other three units has a negligible effect on the external load stresses in the nozzles.

2. The thickness of the nozzle shell course for the Byron Station reactor vessels is 11" minimum, which is somewhat greater than the nozzle shell course thickness of 9-9/16" for the Braidwood Station reactor vessels. The outside radius at the Byron Station inlet nozzle to shell junctions is 6.75" minimum, while the outside radius of the Braidwood Station inlet nozzle to shell junctions is 7.5" minimum. The overall effect of these differences is that the Byron Station nozzle to shell junctions are stiffer than the Braidwood Station nozzle to shell junctions. Therefore, the Braidwood Station nozzle to shell junctions flex a little more, making the primary plus secondary and peak stress intensities in the Braidwood Station nozzles slightly less than the stress intensities in the Byron Station nozzles for corresponding locations.

3. The materials for the Byron Station inlet nozzles are the same as the materials for the Braidwood Station inlet nozzles. The nozzle safe end forgings are all SA-182, F316. The nozzle forgings are all SA-508, Class 2. The nozzle buttering and safe end to nozzle welds are all Ni-Cr-Fe Alloy 600.

4. The same design loadings are applied to the Byron Station and Braidwood Station inlet nozzles. The NSSS design transients are the same for each of the units. The deadweight, thermal, pressure and operating basis earthquake (OBE) seismic loads are the same for each of the units. The design basis steady state operating temperatures and pressures are also the same for each of the units.

Based upon the comparisons in items 1 through 4 above, the calculated stress intensities for the reactor vessel inlet nozzles for the four Byron Station and Braidwood Station Units should be nearly the same. The Byron Station results, as expected, are slightly higher due to the greater stiffness at the nozzle to shell junctions. The maximum ranges of primary plus secondary stress intensities reported in the Byron Station and Braidwood Station stress reports for the thin ends of the nozzles at the beginning of the nozzle tapers reflect this conclusion. The Byron Station value in the original B&W report is 74.7 ksi, vs. the Braidwood Station value of 73.6 ksi.

Similar results are found at the other locations along the nozzles except at the safe ends. At the safe end, the Byron Station report lists the same values for the stress intensity ranges as are reported for the thin end of the nozzle at the beginning of the taper, including the 74.7 ksi maximum value. These values conservatively include the effect of greater stress intensities due to structural discontinuity, even though such discontinuity does not exist at the safe end. The Braidwood Station report, on the other hand, lists the maximum range at the inlet nozzle safe end to be 49.2 ksi (i.e., a lower value because the discontinuity is not assumed).

Both the Byron Station and Braidwood Station maximum ranges of stress intensities are documented in the reactor vessel stress reports. The Braidwood Station result is very near the maximum range of primary plus secondary stress intensity reported for other vessels with similar safe end designs, e.g., Comanche Peak and Watts Bar, which were fabricated by different vessel manufacturers. The results for these other vessels do not exceed the $3S_m$ limit (S_m = Membrane Stress Intensity from the ASME Code) of 51.0 ksi. Based upon the comparison of results with the Braidwood Station vessels and the other similar vessels, it is concluded that the Byron Station maximum range of stress intensity result is conservative by about 25 ksi. An increase in the Byron Station value from 74.7 ksi to 84.5 ksi as a result of the vessel evaluation for the T_{hot} Reduction Program provides additional conservatism. This conservatism is magnified in the fatigue analysis due to the simplified elastic-plastic analysis that was performed to account for exceeding $3S_m$. Therefore, the Byron Station Unit 1 results are conservative and bounding for the other units.

Table 5.2-1 in the Byron and Braidwood Station UFSAR lists ASME Boiler and Pressure Vessel (B&PV) Code, Section III, "Rules for Construction of Nuclear Power Plant Components," 1971 Edition, with addenda through Summer 1973, as the applicable code for the reactor vessel. This is in agreement with the Byron Station and Braidwood Station reactor vessel equipment specifications and reactor vessel stress report certifications. ASME B&PV Code, Section III, 1971 Edition with addenda through Summer 1973 was used in the Byron Station and Braidwood Station reactor vessel evaluation for the power uprate as reported in Section 5.1 of the Power Uprate Licensing Report (LR).

J.2 *In Section 5.2.2.2, the licensee stated that there is sufficient margin to accommodate the increase in fluid induced vibration (FIV) loads for the power uprate. Discuss the potential for FIV with regard to critical fluid elastic stability ratio and the acceptance criteria for the FIV loads for the power uprate. Also, provide a comparison of the maximum FIV load on the most critical component (i.e., guide tubes) at the uprated power level with the allowable FIV load at the Braidwood Station and Byron Station.*

J.2 Response

The current design temperature range between T_{cold} and T_{hot} is 60.0 °F and changes to 66.0 °F with the implementation of power uprate at Byron Station and Braidwood Station.

This power uprate design condition will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces, induced by flow. The corresponding T_{cold} and T_{hot} fluid densities increase by about 2.4%.

Evaluations performed for power uprate conditions show that the flow induced vibration loads on the guide tubes and the upper support columns increase by approximately 2.15% and the impact on the lower internals is approximately a 5% increase. Previous FIV analysis for similar 4-loop reactors has shown sufficient margins. Therefore, the effect on the flow-induced vibration of the reactor internals is considered acceptable for the power uprate at Byron Station and Braidwood Station.

An evaluation of the critical fluid-elastic stability velocity ratio was performed for the most limiting components, which are the upper internals guide tubes, because of their lower natural frequency relative to the support columns. The guide tubes with the maximum flow velocities, at location M-14 and the symmetrically equivalent locations, were studied for the most limiting case. The procedures in Appendix N of the ASME Code, Section III were followed for this evaluation. The changes in the RCS conditions due to power uprate have no significant effect on the guide tube fluid-elastic stability ratio. As a result of this evaluation, the critical stability velocity ratio is 0.86 based on the most limiting uprated conditions. The critical stability velocity ratio is also acceptable throughout heat-up from cold start-up conditions with values comparable or slightly lower than at hot full power conditions.

The calculated stress from FIV, for a plant with similar type internals and the corresponding code allowable, is given below:

Component	Calculated Stress (psi)	Code Allowable (psi)
Core Barrel Beam and Shell Modes – Barrel Nozzle	2,250	13,200
Lower Support Plate	1,625	13,200
Upper Support Plate	4,340	13,200

The amount of margin available is more than sufficient to cover the temperature/fluid density changes seen as a result of power uprate at Byron Station and Braidwood Station.

J.3 *In Section 5.2.3, the licensee evaluated reactor internal components for the uprated power conditions including the lower core plate, baffle/barrel region components, core barrel, baffle plate, baffle/barrel bolts, and upper core plate. In Section 5.2.3.1, you stated that Table 5.2.3-1 lists the evaluation results. However, there are no results found in Table 5.2.3-1 for the reactor internal components mentioned above. Please provide such results including the Code and Code Edition used for evaluation of the reactor internal components. If different from the code of record, please justify and reconcile the differences.*

J.3 Response

Since the Byron Station and Braidwood Station reactor internals were designed prior to the introduction of Subsection NG of the ASME B&PV Code Section III, a plant specific stress report is not required. However, the design of the reactor internals is evaluated according to Westinghouse criteria which are similar to the criteria in Subsection NG of the ASME Code. Hence the acceptance criteria are the same as used in the original design of the plant and its original licensing basis.

Lower Core Plate

New CUF calculations were performed for power uprate program at Byron Station and Braidwood Station. The new CUF calculations reflect the increase in heat generation seen by the lower core plate.

Normal & Upset (Level A & B):

Stress Intensity = 29.188 ksi, allowable $3S_m$ = 51 ksi,

Faulted (Level D):
P_m, P_L and $P_m + P_b$ are unaffected by power uprate
(P_m = general primary membrane stress intensity; P_L = local primary stress intensity; P_b = primary bending stress intensity)

Fatigue:
S_{alt} = 17 ksi, allowable number of cycles 10^7, calculated cumulative usage factor (CUF) = 0.0 (Allowable CUF=1.0)
(S_{alt} = alternating stress)

Baffle-Barrel Region Components

No new CUF calculations were performed for baffle-barrel region components (core barrel, baffle plates, bolting and former plates). The effect of heat generation rates seen by these components due to power uprate conditions is bounded by the current analysis of record. Therefore, the current analysis of record remains applicable for the power uprate conditions at Byron Station and Braidwood Station.

Upper Core Plate

No new CUF calculations were performed. The effect of heat generation rates and the new design transients on the upper core plate were negligible. The current analysis of record remains bounding for the power uprate conditions at Byron Station and Braidwood Station.

J.4 *In regard to Sections 5.4 of Attachment E to the reference transmittal, please provide the maximum calculated stress and CUF at the critical locations of the control rod drive mechanisms (CRDM's) components. Also, provide the allowable code limits, and the Code and Code Edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.*

J.4 Response

The maximum stress intensities for all of the analyses, with the possible exception of the fatigue analyses, remain bounding for power uprate (LR Section 5.4.3.2). The upper joint canopy of the CRDMs has been previously shown to be the area of highest fatigue usage. Since none of the conditions associated with power uprate would cause a change in the location of the area of highest fatigue usage, the upper joint canopy was again evaluated for fatigue.

As stated in LR Section 5.4.4, "The total change in fatigue usage for the power uprate is +0.007. The previous analysis worst-case fatigue usage for the upper joint canopy was 0.934. The worst-case fatigue usage for the power uprating is 0.941 for this same upper joint canopy. This is still less than the Code allowable of 1.0. Therefore, the CRDMs continue to meet the E-specifications and the ASME Code of record for uprate program."

For the upper joint canopy, the calculated stress was 40,057 psi versus an allowable of 48,300 psi.

The code of record is the ASME B&PV Code, Section III, 1974 Edition with Addenda through Summer 1974. This was the original Code of Record and has not changed due to power uprate.

J.5 *In reference to Section 5.5, the licensee stated that the system design transients are used in the evaluation of the piping fatigue, and that the impact of changes in the system thermal transients were factored into the ASME code stress and CUF determination. Discuss in detail the methodologies and assumptions for the determination of the ASME Code stress and CUF for reactor coolant loop piping and branch piping. Provide the maximum calculated stresses and CUFs at the most critical locations for reactor coolant loop (RCL) piping, primary equipment supports and nozzles, RCS branch nozzles and pressurizer surge nozzles, allowable limits, the Code of record and Code Edition used for the power uprate conditions for NSSS piping and supports. If the Code Edition used was different from the Code of record, justify and reconcile the differences.*

J.5 Response

The methodologies and assumptions used in evaluating the impact of power uprate on the Reactor Coolant Loop analysis for Byron Station and Braidwood Station are consistent with the original analysis methodology provided in the piping analysis design specification and described in the UFSAR. A discussion of the methodologies and assumptions used in the power uprate evaluation follows.

Design and Faulted Condition Stress Evaluation

The following analysis inputs remain unchanged for the design and faulted condition stress evaluation because they are either unaffected or remain bounding.

- The weight of the piping systems
- The seismic response spectra
- The existing loss of coolant accident (LOCA) loop forces
- The existing LOCA vessel motions

Since piping weight, response spectra, LOCA forces and reactor vessel displacements remain unchanged, there is no change to the deadweight, seismic or LOCA analyses of record for Byron Station and Braidwood Station.

The stresses due to pressure, weight, and OBE for the design condition, and pressure, weight, safe shutdown earthquake (SSE), and LOCA for the faulted condition are combined as specified in the original design specification using Equation 9 of subsection NB-3650 of the ASME Code. Analysis of record design, faulted, and allowable stresses that are representative for all four units and remain applicable after power uprate, are presented in Tables J.5-1 and J.5-2 for the reactor coolant loop and branch line nozzles.

Fatigue Evaluation

The thermal moment stress from temperature changes to the hot leg and cold leg fluid were shown to be insignificant. The data below shows the existing and post-uprate temperatures.

	Existing		Post Uprate	
	High T_{ave}	Low T_{ave}	High T_{ave}	Low T_{ave}
Hot Leg (°F)	618.4	600.0	618.4	608.0
Cold Leg (°F)	558.4	538.2	555.7	538.2

Thermal transients and number of cycles either remained unchanged or were shown to be insignificant changes from those considered in the original design.

Since thermal moment stress remains unchanged and there are insignificant changes to thermal transients and cycles, there are no changes to the fatigue analysis of record for Byron Station and Braidwood Station (see the response to Question J.6 for discussion of the impact of steam generator replacement on piping stresses).

The original stresses due to pressure, weight, thermal and OBE for the fatigue stress combinations were evaluated in accordance with the piping design specification and the rules of Subsection NB-3653 of the ASME Code, and they are unchanged for power uprate. ASME Subsection NB-3653 Equations 12 and 13 stresses and usage factors for fatigue are presented in Table J.5-1 for the reactor coolant loop and in Table J.5-2 for the branch line nozzles.

ASME Codes Used for Power Uprate

The ASME codes used for the power uprate evaluations are currently applicable to both Byron Station and Braidwood Station and are listed in the UFSAR. The codes utilized are as follows.

1. 1974 Edition of the ASME code, including Summer 1975 Addenda

2. 1977 Edition and addenda up to and including Summer 1979 for stress analysis purposes for the reactor coolant primary loop piping and branch nozzles, as specified in the design specification, except as modified below.

    - Design, material supply, examinations and welding of the hot leg and cold leg fast response RTD thermowells, crossover leg nozzle caps and installation bosses meet the requirements of the 1983 Edition of the ASME code. The concerns of low and high cycle fatigue were also evaluated using the fatigue curves of the 1986 version of the code.

- The 1986 edition of the code was used to evaluate fatigue on surge lines with stratification loading. This was based on the requirement of NRC Bulletin 88-11 to use the "latest ASME III requirements incorporating high cycle fatigue".

3. ASME B&PV Code, Section I, Subsection NA, Appendices I, XVII and F, 1974 Edition with Addenda through Summer 1975; ASME B&PV Code, Section III, Subsection NF, 1974 Edition with Addenda through Summer 1975; and AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (for supports)

Primary Equipment Support Evaluation

The parameters affected by power uprate have an insignificant impact on support loading since the maximum hot leg (HL) and cold leg (CL) temperatures do not increase (i.e., HL_{tave} is 618.4°F before and after uprate and CL_{tave} decreases from 558.4°F to 555.7°F after uprate). Therefore, the support evaluations performed in the original analyses remain valid. The existing maximum support interactions for Byron Station and Braidwood Station are provided in Tables J.5-3 through J.5-6.

Primary Equipment Nozzle Evaluation

The original piping analysis loads are compared to equipment allowable loads to show qualification. The parameters affected by uprate have an insignificant impact on the primary equipment nozzle loads since the maximum hot leg and cold leg temperatures do not increase (HL_{tave} is 618.4°F before and after power uprate and CL_{tave} decreases from 558.4°F to 555.7°F after uprate). Therefore, the primary equipment nozzle evaluations performed in the original analysis remain valid.

References for Question Set J.5

1. Calculation No. 5.11-BYR97-252, Revision 0, Byron Station Unit 2, "Stress Report for NSSS Component Supports"

2. Calculation No. 5.11-BRW-97-0570, Revision 0, Braidwood Station Unit 2, "Stress Report for NSSS Component Supports"

Table J.5-1
ASME Code Reactor Coolant Loop Stress (psi)
Representative of
Byron Station and Braidwood Station Units 1&2

	Eq. 9 Design	Allowable Stress	Eq. 9 Faulted	Allowable Stress	Eq. 12	Allowable Stress	Eq. 13	Allowable Stress	Usage Factor
Hot Leg (1), (2)	20610	26400	33800	52800	18840	52800	44120	52800	0.8733
X-over Leg (3), (4), (5)	21270	26400	38910	52800	7070	52800	42010	52800	0.200
Cold Leg (6), (7)	20230	26400	46070	52800	8260	52800	42400	52800	0.300

Notes:
1. Maximum equation 9 design, faulted and equation 12 stress occurs at the 50 degree elbow.
2. Maximum equation 13 stress and usage factor occurs at the reactor pressure vessel outlet nozzle weld.
3. Maximum equation 9 design and faulted stress occurs at the 40 degree elbow.
4. Maximum equation 12 stress occurs at the 90 degree elbow.
5. Maximum equation 13 stress and usage factor occurs at the steam generator outlet nozzle weld.
6. Maximum equation 9 design, faulted and equation 12 stress occur at the 22 degree elbow.
7. Maximum equation 13 stress and usage factor occurs at the reactor pressure vessel inlet nozzle weld.

Table J.5-2
ASME Code Branch Nozzle Stress (psi)
Representative of
Byron Station and Braidwood Station Units 1&2

Branch Nozzle	Eq. 9 Design	Allowable Stress	Eq. 9 Faulted	Allowable Stress	Eq. 12	Allowable Stress	Eq. 13	Allowable Stress	Usage Factor
1-inch RTD Thermowell – H.L.	13900	24300	31600	48600	23000	48600	44060	48600	0.960
2-inch RTD Thermowell and Boss – C.L.	17360	24300	39800	48600	28500	48600	49900	50520	0.950
3-inch RTD Return Cap – X.O.L.	11290	24300	16500	48600	18750	48600	44000	48600	0.950
4-inch Pressurizer Spray – C.L.	9600	24300	26600	48600	11200	48600	32500	48600	0.400
14-inch Pressurizer Surge – H.L.	10500	25000	30500	50100	21000	61500	54450	61500	0.982
2-inch Loop Fill – X.O.L.	9743	24300	21684	48600	39000	48600	46200	48600	0.400
¾ - inch Sample – H.L.	15130	24300	22500	48600	7620	48600	42800	48600	0.980
3-inch Loop Stop Valve Bypass Line – C.L.	9500	24300	27800	48600	4200	48600	31800	48600	0.200
2-inch Loop Drain – X.O.L.	9540	24300	24600	48600	35200	48600	37900	48600	0.470
2-inch Excess Letdn/Drain – X.O.L.	9380	24300	31080	48600	39400	48600	41500	48600	0.200
3-inch Normal Letdown – X.O.L.	9330	24300	28610	48600	21200	48600	33110	48600	0.200
3-inch Normal/Alt. Charging – C.L.	10000	24300	31100	48600	30000	48600	43230	48600	0.750
6-inch SIS Nozzle – H.L.	9710	24300	38040	48600	4000	48600	32400	48600	0.010
3-inch Boron Injection – C.L.	9110	24300	16600	48600	6000	48600	55400	56700	0.900
10-inch Accumulator – C.L.	25800	28400	44300	56700	25000	48300	69200	69250	0.950
12-inch RHR – H.L.	10550	24300	26000	48600	12000	48600	41000	48600	0.085

Table J.5-3
Reactor Pressure Vessel Supports

Byron Station Unit 2 [1]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Anchor Bolts	16.5	45.6	36
Normal Stresses in Stiffeners (Membrane)	35.5	49	72
Normal Stresses in Stiffeners (Membrane + Bending)	41.6	73.5	57
Normal Stresses in Webs	44	49	90

Braidwood Station Unit 2 [2]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Anchor Bolts	16.5	45.6	36
Normal Stresses in Stiffeners (Membrane)	31.8	49	65
Normal Stresses in Stiffeners (Membrane + Bending)	64.8	73.5	88
Normal Stresses in Webs	46	49	94

Table J.5-4
Steam Generator Supports

Byron Station Unit 2 [1]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Normal Stresses in Inner Frame of Lower Lateral Support	38.6	94.5	41
Tensile Stresses in Bolts of Columns	63.9	81.9	78
Tensile Stresses in Through Thickness Direction of Plates for Columns	11.1	24.5	45
Tensile Stresses in Bolts of Upper Lateral Support	54.2	81.3	67
Maximum Stresses for Upper Lateral Supports	31.1	49	63

Maximum Interaction Ratio for Combined Stresses [1]	
Component	Interaction Ratio Shown as a Percentage
Outer Frames of Lower Lateral Supports	98
Columns	84
Upper Lateral Support Bolts	83

Braidwood Station Unit 2 [2]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Normal Stresses in Inner Frames of Lower Lateral Support	47.9	94.5	51
Tensile Stresses in Bolts of Columns	63.9	81.9	78
Tensile Stresses in Through Thickness Direction of Plates for Columns	11.1	24.5	45
Tensile Stresses in Bolts of Upper Lateral Support	54.2	81.3	67
Maximum Stresses for Upper Lateral Supports	31.1	49	63

Maximum Interaction Ratio for Combined Stresses [2]	
Component	Interaction Ratio Shown as a Percentage
Outer Frames of Lower Lateral Supports	99
Columns	84
Upper Lateral Support Bolts	83

Table J.5-5
Reactor Coolant Pump Supports

Byron Station Unit 2 [1]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts of Columns	86.4	100.3	86
Tensile Stresses in Through Thickness Direction of Plates for Columns	17.1	24.5	70

Maximum Interaction Ratio for Combined Stresses [1]	
Component	Interaction Ratio Shown as a Percentage
Lateral Supports	99
Columns	92

Braidwood Station Unit 2 [2]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts of Columns	86.4	100.3	86
Tensile Stresses in Through Thickness Direction of Plates for Columns	17.1	24.5	70

Maximum Interaction Ratio for Combined Stresses [2]	
Component	Interaction Ratio Shown as a Percentage
Lateral Supports	99
Columns	92

Table J.5-6
Pressurizer Supports

Byron Station Unit 2 [1]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts for Lower Lateral Supports	76.7	93.9	82

Maximum Interaction Ratio for Combined Stresses [1]	
Component	Interaction Ratio Shown as a Percentage
Lower Lateral Support	87
Upper Lateral Support	54

Braidwood Station Unit 2 [2]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts for Lower Lateral Supports	76.7	93.9	82

Maximum Interaction Ratio for Combined Stresses [1]	
Component	Interaction Ratio Shown as a Percentage
Lower Lateral Support	87
Upper Lateral Support	54

J.6 *In reference to Section 5.5, the licensee stated that the current analysis for the reactor loop primary piping was performed by Framatome Technologies, Incorporated (FTI) for Byron Station and Braidwood Station, Units 1 due to steam generator replacement at Units 1. The FTI analysis is also used to demonstrate primary equipment supports to be adequate for the power uprate. It is noted that the FTI analysis has not been reviewed by the NRC. Please describe the analysis method, assumptions, computer codes used for analysis (if different from those specified in the UFSAR), the results of the analysis, and the Code and Code Edition used. If Code Edition used was different than the Code of record, justify and reconcile the differences. The results of analyses should include RCL piping, major equipment supports and nozzles, RCL branch nozzles and pressurizer surge nozzles.*

<u>J.6 Response</u>

The methodology and assumptions used for the RCL piping due to steam generator replacement for Byron Station Unit 1 and Braidwood Station Unit 1 were not altered from the methodology and assumptions used in the original design basis analyses performed by Westinghouse. There are no changes to the Codes shown in the response to Question J.5 as a result of the steam generator replacement analyses [1].

A four loop model was created by FTI for Byron Station Unit 1 and Braidwood Station Unit 1 using FTI finite element and post processor programs (i.e., BWSPAN and BWSPEC respectively) [2]. This model couples the primary equipment support elements with the interior concrete structure (ICS) to eliminate excess conservatism previously included in the Westinghouse model [1]. The analysis was performed using the leak before break (LBB) design basis, which eliminates the effects of RCS pipe breaks. In addition, jet impingement resulting from arbitrary intermediate breaks (AIBs) on surrounding high energy lines were removed from the analysis for eliminated AIBs.

The analysis results from the model were benchmarked in its pre-modified configuration against the original Westinghouse loop model. The post-modified analysis results were then compared to the existing Westinghouse design basis Westinghouse analysis results [1]. This comparison was used to justify the applicability of the existing design basis results for Code stresses/CUFs. Justification of any minor differences between the results of the FTI analysis and the existing design basis was provided in the supporting calculation.

The calculations performed by FTI demonstrate that the stresses, usage factors, and allowables from the existing design basis analysis by Westinghouse remain valid for the RCL piping. Therefore, the pipe stresses/CUFs as reported in the response to Question J.5 (i.e., Table J.5-1) remain valid.

The existing maximum interactions for critical members of the reactor pressure vessel (RPV), steam generator (SG), and reactor coolant pump (RCP) supports due to steam generator replacement on Byron Station Unit 1 and Braidwood Station Unit 1 are shown in the tables J.6-1 through J.6-4 . These remain valid for power uprate conditions.

Table J.6-5 provides the ratios of the maximum force and moment to allowable interaction (IC) of the RPV, SG and RCP primary nozzles for OBE, SSE, LOCA, and high energy line break (HELB) [3]. Weight and thermal expansion loading on these nozzles are not provided since they were not changed by steam generator replacement [2]. Unique analyses were performed for Byron Station Unit 1 and Braidwood Station Unit 1 since two snubbers were eliminated from each steam generator's upper lateral supports at Byron Unit 1, resulting in a reduction in support stiffness [1].

The calculations performed by FTI demonstrate that the RCL branch nozzle loads and allowables from the existing design basis analysis by Westinghouse remain valid. Therefore, the pipe stresses/CUFs as reported in the response to Question J.5 (i.e., Table J.5-2) remain valid.

References for Question Set J.6:

1. Framatome Technologies, Inc. (FTI) Calculation 51-1240322, Revision 2, "AIS for Structural Analysis of the Primary Loop with the RSGs"

2. Framatome Technologies, Inc. (FTI) Calculation 32-1236230, Revision 4, "Byron 1 and Braidwood 1 Loop Anal. w/ Replacement Steam Gen."

3. Framatome Technologies, Inc. (FTI) Calculation 51-1239757, Revision 1, "Loading Specification for Steam Generator Replacement, Byron/Braidwood Unit 1"

4. Calculation No. 5.11-BYR97-251, Revision 0, Byron Station Unit 1, "Stress Report for NSSS Component Supports"

5. Calculation No. 5.11-BRW-97-0569, Revision 0, Braidwood Station Unit 1, "Stress Report for NSSS Component Supports"

Table J.6-1
Reactor Pressure Vessel Supports

Byron Station Unit 1 [4]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Anchor Bolts	11.9	45.6	26
Normal Stresses in Stiffeners (Membrane)	23	49	47
Normal Stresses in Stiffeners (Membrane + Bending)	46.7	73.5	63.5
Normal Stresses in Webs	33.3	49	68

Braidwood Station Unit 1 [5]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Anchor Bolts	13.7	45.6	30
Normal Stresses in Stiffeners (Membrane)	26.6	49	54.2
Normal Stresses in Stiffeners (Membrane + Bending)	53.9	73.5	73.3
Normal Stresses in Webs	38.5	49	78.4

Table J.6-2
Steam Generator Supports

Byron Station Unit 1 [4]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Normal Stresses in Stiffeners of Inner Frames of Lower Lateral Support	46.56	73.5	63.3
Normal Stresses in Inner Frame of Lower Lateral Supports	57.3	73.5	78
Tensile Stresses in Bolts of Columns	63.9	81.9	78
Tensile Stresses in Through Thickness Direction of Plates for Columns	9.51	24.5	39
Tensile Stresses in Bolts of Upper Lateral Support	42.74	81.3	53
Maximum Stresses for Upper Lateral Supports	36.34	41.3	88

Maximum Interaction Ratio for Combined Stresses [4]	
Component	Interaction Ratio Shown as a Percentage
Outer Frames of Lower Lateral Supports	97.7
Columns	66.2
Upper Lateral Support Bolts	83
Snubber End Block	78

Braidwood Station Unit 1 [5]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Normal Stresses in Inner Frames of Lower Lateral Support	51.9	94.5	54.9
Tensile Stresses in Bolts of Columns	63.9	81.9	78
Tensile Stresses in Through Thickness Direction of Plates for Columns	11.02	24.5	45
Tensile Stresses in Bolts of Upper Lateral Support	42.5	81.3	52.3
Maximum Stresses for Upper Lateral Supports	36.34	41.3	88

Maximum Interaction Ratio for Combined Stresses [5]	
Component	Interaction Ratio Shown as a Percentage
Outer Frames of Lower Lateral Supports	98.1
Columns	71.4
Upper Lateral Support Bolts	91.4

Table J.6-3
Reactor Coolant Pump Supports

Byron Station Unit 1 [4]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts of Columns	19.94	100.3	20
Tensile Stresses in Through Thickness Direction of Plates for Columns	4.21	24.5	17.2

Maximum Interaction Ratio for Combined Stresses [4]	
Component	Interaction Ratio Shown as a Percentage
Lateral Supports	71.3
Columns	35.1

Braidwood Station Unit 1 [5]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts of Columns	23.57	100.3	23.6
Tensile Stresses in Through Thickness Direction of Plates for Columns	4.97	24.5	20.3

Maximum Interaction Ratio for Combined Stresses [5]	
Component	Interaction Ratio Shown as a Percentage
Lateral Supports	80
Columns	40

Table J.6-4
Pressurizer Supports (unaffected by steam generator replacement)

Byron Station Unit 1 [4]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts for Lower Lateral Support	76.7	93.9	82

Maximum Interaction Ratio for Combined Stresses [4]	
Component	Interaction Ratio Shown as a Percentage
Lower Lateral Support	87
Upper Lateral Support	54

Braidwood Station Unit 1 [5]

Component	Maximum Calculated Stress (ksi)	Allowable Stress (ksi)	Percentage of Max. Stress to Allowable
Tensile Stresses in Bolts for Lower Lateral Supports	76.7	93.9	82

Maximum Interaction Ratio for Combined Stresses [5]	
Component	Interaction Ratio Shown as a Percentage
Lower Lateral Support	87
Upper Lateral Support	54

Table J.6-5
Primary Equipment Nozzles
Byron Station Unit 1

Nozzle/IC	OBE		SSE		LOCA / HELB*	
	Force	Moment	Force	Moment	Force	Moment
RPV Inlet	0.305	0.124	0.362	0.204	0.537	0.354
RPV Outlet	0.257	0.317	0.476	0.291	0.619	0.414
SG Inlet	0.213	0.120	0.386	0.251	0.480	0.726
SG Outlet	0.147	0.166	0.247	0.248	0.168	0.072
RCP Suction	0.373	0.163	0.417	0.266	0.553	0.093
RCP Discharge	0.192	0.294	0.178	0.297	0.264	0.071

Braidwood Station Unit 1

Nozzle/IC	OBE		SSE		LOCA / HELB*	
	Force	Moment	Force	Moment	Force	Moment
RPV Inlet	0.334	0.154	0.488	0.252	0.537	0.354
RPV Outlet	0.416	0.447	0.882	0.354	0.619	0.415
SG Inlet	0.340	0.198	0.701	0.500	0.480	0.726
SG Outlet	0.174	0.204	0.353	0.337	0.168	0.071
RCP Suction	0.419	0.199	0.555	0.426	0.553	0.093
RCP Discharge	0.210	0.326	0.223	0.364	0.264	0.071

*load is combined with WEIGHT for RPV nozzle evaluation.

J.7 *In Section 5.7.1, the current design basis input parameters for the Unit 1 RCL piping analysis by FTI are provided in Table 5.7.1.1-1 where the steam flow for the 5 percent power uprate is $4.0x10^6$ lbm/hr. This is different from Table 2.1-1 where the steam flow is $15.98x10^6$ lbm/hr. Please explain the apparent discrepancy.*

J.7 Response

Section 5.7.1 discusses the design parameters of a Replacement Steam Generator (RSG). The value provided in Table 5.7.1.1-1 (i.e., $4.0X10^6$ lb_m/hr) is the steam flow from a single RSG. LR Table 2.1-1 provides design data for Unit 1, thus the second value (i.e., $15.96X10^6$ lb_m/hr) represents the steam flow from all four RSGs in Unit 1.

J.8 *In Section 5.7.1.5, the licensee indicated that the potential for vibration in the U-bends (including the small radius) and tubes due to fluid elastic instability at the power uprate conditions was assessed. Provide an evaluation of the flow-induced vibration of the steam generator U-bend tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated elastic-fluid instability ratio. If any computer codes used in the analysis are different from those specified in the Braidwood Station and Byron Station FSAR, provide the basis for the acceptability of the computer codes that were used.*

J.8 Response

The evaluation of the flow-induced vibration of the steam generator U-tubes due to power uprate is documented in Table 4B of the Babcock & Wilcox (B&W) "Flow-Induced Vibration Analysis Report," B&W-222-7720-FIV-01, April 2000, Revision 2.

The FIV analysis at 100% power identified that the worst case for flow-induced vibration is U-bend Tube MK#116-Row #1 for the case of corroded conditions where a corrosion allowance of 0.00025 inches has been applied to the primary side (i.e., Tube ID) and a corrosion allowance of 0.00055 inches has been applied to the secondary side (i.e., Tube OD). A flow-induced vibration analysis was done for this worst case at 105% power based on the homogenous flow velocities and densities as calculated by the 3D Code ATHOSBWI.

The U-bend region is modelled from the seventh lattice grid to the top of the U-tube on both the hot and cold legs. The lattice grid and U-bend supports are modelled as "pinned but axial free", with the exception of the lowest lattice grid which is "pinned". These support conditions are for the normal "wet supports". The U-bend region is also modelled for crudded supports where the tube becomes frozen or clamped at the support. The crudded support is considered as excessively conservative since fully crudded locked supports are not expected with the lattice grid and U-bend flatbar support components utilized by B&W.

The first twenty natural frequencies and mode shapes for the tube model are obtained from MSC/pal 2, a finite element macro accessed within the program

EasyFIV. Damping values are calculated for 105% power operation based on an RCS T_{hot} temperature cases of 608°F and 618.4°F.

The EasyFIV program determines the tube vibration response due to:

a) Fluid elastic Instability
b) Vortex Shedding Resonance
c) Random Turbulence Excitation

The EasyFIV program uses the empirical expression developed by Connors for the critical velocity to cause the onset of fluid-elastic instability. The fluid-elastic results are presented in terms of U_{eff}/U_{cr}, which is the ratio of the effective fluid velocity (i.e., average in the gap between tubes), to the critical velocity at the onset of fluid elastic instability.

As reported in Table 4B, the worst case critical velocity ratio is 0.772 for wetted supports and 0.985 for crudded supports. This occurs at the lower end of the T_{hot} window when the RCS T_{hot} is 608°F. Normally the unit will operate at a higher T_{hot} and the critical velocity ratio will be lower. The maximum vibration level due to random turbulence excitation is 0.470 root mean square (RMS) mils. Due to the two-phase flow, vortex shedding cannot occur in the U-bend region.

The B&W personal computer based code EasyFIV, Version 3.3, was used for the flow induced vibration analysis. The verification package, which describes the applicability and limitations of the program, resides in the B&W Computer Applications quality assurance file. This computer code is listed in the Byron Station and Braidwood Station UFSAR for this application.

J.9 *Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate regarding the determination that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety–related motor operated valves (MOVs) in the generic letter 89-10 MOV program at Braidwood Station and Byron Station will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level could not be confirmed. Please discuss the effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power–operated gate valves per GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power–Operated Gate Valves," and on the evaluation of over pressurization of isolated piping segments per GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions."*

J.9 Response

The safety-related system and component functionality and evaluations performed to ensure safety-related systems and components meet their performance and technical specifications are addressed in Licensing Report

Section 4.0 "NSSS Systems" and Section 9.0 "BOP Systems, Structures and Components".

Safety-related system and component performance and technical specification requirements were not changed for the power uprate. The safety-related systems and components were evaluated to demonstrate their existing performance and capacity are adequate to perform their safety-related functions at the uprated power level. No safety-related systems or components required modification to meet their safety-related performance requirements at the uprated power level.

The safety-related system and component operating conditions (i.e., pressure, temperature, and flow) for systems where MOVs are included in Generic Letter (GL) 89-10 were not adversely impacted by the power uprate. The impact of increased operating parameters on the design basis differential pressures used in the GL 89-10 program was evaluated. The increased flow requirements in some safety-related systems, due to power uprate, will increase the differential pressures across the associated MOVs. The Byron Station and Braidwood Station evaluations performed in response to GL 89-10 were conservatively based on pump shutoff head, relief and safety valve setpoints plus accumulation, containment design pressure, and interlock setpoints which are not changed as a result of power uprate. Therefore, power uprate does not impact the GL 89-10 Program.

The revised post-accident temperature and pressure conditions for systems and components that are subject to pressure locking and thermal binding were not impacted; therefore, power uprate does not impact the GL 95-07 evaluations. The revised post accident temperature and pressure conditions for systems and components that are subject to overpressurization of isolated piping, and the post-accident containment atmosphere and pressure were not increased as documented in the power uprate analyses and, therefore, power uprate does not impact the GL 96-06 evaluations.

J.10 *In reference to Section 9, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchorage for pipe supports. Provide the maximum calculated stresses for the critical BOP piping systems, the allowable limits, the Code of record and the Code Edition used for the power uprate conditions. If the Code Edition used was different from the Code of record, justify and reconcile the differences. Were the analytical computer codes used in the evaluation different from those used in the original design basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.*

<u>J.10 Response</u>

The BOP piping systems evaluations are discussed in LR Section 9.3.20. Operation at the power uprate conditions may increase piping stresses caused by slightly higher operating temperatures, pressures and flow rates. Additionally, BOP pipe supports and equipment nozzles may be potentially subjected to slightly increased loadings due to power uprate.

The safety-related piping systems evaluated for power uprate are as follows.

- Auxiliary Feedwater
- Chemical and Volume Control
- Component Cooling Water
- Containment Spray
- Essential Service Water
- Feedwater
- Fuel Pool Cooling
- Main Steam
- Residual Heat Removal
- Reactor Coolant
- Reactor Coolant Sampling
- Safety Injection
- Steam Generator Blowdown

The non safety-related piping systems evaluated for power uprate are as follows.

- Auxiliary Steam
- Chilled Water
- Circulating Water
- Condensate
- Condensate Booster
- Condenser Off Gas
- Extraction Steam
- Heater Drains
- Non-Essential Service Water

The piping system evaluations for power uprate were performed by determining "change factors" for the changes in thermal, pressure, and flow rate conditions.

The thermal "change factor" was based on the ratio of the thermal power uprate to pre-thermal power uprate operating temperature. That is, the thermal change factor is $(T_{uprate}-70°F)/(T_{pre-uprate}-70°F)$. Using this method for the thermal change factor, evaluations resulted in a reasonable approximation of the thermal impact on piping stresses and loads. For piping which experiences less than a 10°F increase, the calculated change factors are virtually identical to those determined using the applicable mean coefficients of expansion for a temperature increase of < 10°F.

The pressure "change factor" was determined by the $P_{uprate}/P_{pre-uprate}$ ratio. For the main steam and feedwater piping systems which experience only higher fluid flow rates for power uprate conditions, a separate flowrate assessment was performed in their respective piping system evaluations. The flow rate "change factor" was determined by the flow rate $_{uprate}$/flow rate $_{pre-uprate}$ ratio.

Based on the thermal, pressure, and flow rate change factors determined, the following engineering activities were performed or conclusions reached:

- For thermal, pressure, and flow rate change factors less than or equal to 1.0 (i.e., the pre-thermal power uprate condition bounds or equals the thermal power uprate condition), no further review was required and the piping system was concluded to be acceptable for the thermal power uprated conditions.

- Thermal, pressure, and flow rate change factors of 1.0 through 1.05 (i.e. up to a five percent increase in thermal expansion and/or pressure stress effects), were concluded to be acceptable by engineering judgement since these increases are offset by conservatisms in analytical methods used to calculate the existing thermal stresses and loads. Conservatisms include the enveloping of multiple thermal operating conditions and not considering pipe support gaps in the thermal analyses. Pressure effects are always considered in conjunction with other loading conditions (e.g., weight and seismic), thus the overall effect of the pressure change factor is reduced.

- Thermal, pressure, and flow rate change factors of 1.05 through 1.10 (i.e., up to a ten percent increase in thermal expansion and/or pressure and/or flow rate stress effects) are acceptable based on historical evaluations and case studies of conservatism in the analytical techniques used to predict dead weight, thermal, fluid transient, and seismic loads. For seismic loads, the approach is limited to systems previously analyzed using linear elastic models and the enveloped response spectrum (ERS) method of analysis. In addition, the effects of the Pressure Vessel Research Committee (PVRC) damping and support stiffness are considered.

- Thermal, pressure, and flow rate change factors of 1.10 to 1.20 (i.e., from ten to twenty percent increases in thermal expansion and/or pressure and/or flow rate stress effects) were individually evaluated to confirm acceptability. No instance of thermal, pressure, or flow rate change factors of 1.20 or greater were identified as a result of power uprate.

The results of the piping systems reviews for the systems identified above are documented by calculation and are summarized below.

- All piping systems affected by the power uprate were determined to have a thermal "change factor" of 1.05 or smaller, or a maximum temperature less than 150°F and an operating temperature range of less than 80°F, except for the Essential Service Water outlet to the Component Cooling Water heat exchanger which experienced a 1.18 thermal change factor during cooldown conditions. This system was specifically evaluated and found to be acceptable. Based on the acceptance criteria described above, all the piping systems were found to be acceptable.

- All piping systems affected by the power uprate were determined to have an operating pressure "change factor" of 1.08 or smaller. Based on the acceptance criteria described above, these piping systems were concluded to be acceptable.

- The main steam and feedwater piping systems were determined to have flowrate "change factors" of 1.08 or smaller. Based on the acceptance criteria described above, these piping systems were concluded to be acceptable.

The piping systems review concluded that all piping systems remain acceptable and will continue to satisfy existing design basis requirements under uprate conditions in accordance with the Code of record, ASME Section III 1974 Edition up to Summer 1975 Addenda and American National Standards Institute (ANSI) B31.1, 1973 Edition, as applicable. No new computer codes were used in the evaluation. The evaluations document that no new or revised stress analyses are required and no piping or pipe support modifications are necessary as a result of the increased power level.

J.11 *Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement, and pipewhip loads for the power uprate condition.*

J.11 Response

The flows through safety related heat exchangers were not revised for power uprate and, therefore, flow induced vibration in safety related heat exchangers was not evaluated.

As part of the review of non-safety related heat exchangers, the design flows were reviewed to ensure that they bounded the flow requirements at power uprate conditions. The flow through the hydrogen cooler for the main generator was evaluated as acceptable when limited to below the design flow value. The main condenser was also evaluated and determined to be acceptable without modification.

The HELB review was conducted to evaluate the possible effects of power uprate on inputs to equipment qualification (EQ) analyses (i.e., pressure, temperature, and flooding), and on the dynamic effects of pipe breaks.

The Byron Station and Braidwood Station HELB analyses are discussed in UFSAR Sections 3.6, "Environmental Design of Mechanical and Electrical Equipment" and 3.11, " Protection Against Dynamic Effects Associated with the Postulated Break of Piping". UFSAR Table 3.6-2 lists the systems previously identified as containing high energy lines inside and outside containment for which breaks must be considered.

This uprate impact review considered the consequences of postulated breaks inside and outside the containment for the following piping systems.

- Reactor Coolant System
- Main Steam System
- Main Feedwater System
- Auxiliary Steam System
- Steam Generator Blowdown System
- Chemical and Volume Control System

The dynamic effects of pipe breaks have typically been determined using the system design pressure. To determine the impact of power uprate on the design basis, a comparison was made of the appropriate design pressure and the expected pressures associated with power uprate.

The resulting conditions (i.e., pressure, temperature, etc.), assuming the postulated failure of the affected piping systems, are acceptable for uprate provided they are bounded by those used in the existing design basis. The design basis conditions are considered bounding if the internal pipe operating conditions used in the HELB analysis of record are bounded by the same operating modes at the power uprate conditions.

For the Main Steam and Main Feedwater Systems, the changes in system pressure due to uprated operations result in pressure values that are below the design basis value (i.e., system design pressure). Therefore, there is no effect on the HELB analysis of record with respect to dynamic effects of pipe breaks. There are no changes in the parameters governing the dynamic effects of HELB for the Auxiliary Steam System, Steam Generator Blowdown System, and Chemical and Volume Control System.

The environmental effects of pipe breaks for equipment and systems which are not subjected to jet impingement consist of room atmosphere temperature, humidity, and flooding. The environment in a compartment or room is determined using system temperature and pressure and the room pressure following the break. A comparison of the system enthalpy, temperature, and pressure (i.e., the basis for the environmental effects) with the uprated values was made to assess the impact of power uprate.

The containment temperature profile and peak pressure used for equipment qualification were compared to the calculated containment temperatures and pressures for power uprate. In addition, the existing design basis subcompartment pressures and temperatures were compared to those calculated for the Main Steam Line Break (MSLB) outside containment.

The containment peak temperatures and pressures following the postulated LOCA and MSLB were reanalyzed by Westinghouse and remain bounded by the original analysis. Although the profiles were changed slightly, the equipment qualification is not impacted. HELB outside containment also does not create environmental conditions that result in challenges to the integrity of safety-related equipment, as discussed in LR 9.3.21.

In summary, the power uprate will not change the temperature and pressure environment used as the basis for equipment qualification. The existing HELB analyses for the Main Steam, Main Feedwater, Auxiliary Feedwater, Steam Generator Blowdown, and Chemical and Volume Control Systems, Auxiliary Steam System, and Steam Generator Blowdown System are not affected by the uprate. The uprate parameters are bounded by the original HELB analysis and no additional analysis is required. Also, no HELB locations are changed due to changes in operating conditions.

J.12 *In Section 9.5.3, the licensee stated that the maximum temperature of 162.7°F (> 150°F allowed by [American Concrete Institute] ACI 349-97 Code) occurs in the spent fuel pool (SFP) for a full off load, and that this is acceptable since the full off load is a temporary condition and the temperature for the long term remains below 150°F. Provide a discussion regarding an evaluation of the SFP including analysis methodology, assumptions, computer codes used for the analysis, and the results of the analysis (i.e., stresses), and allowable limits specified in the ACI 349 Code for application to the uprated power condition. If the computer codes used are different from the codes of record, justify and reconcile the differences.*

J.12 Response

A discussion regarding the SFP evaluation for power uprate, including analysis methodology, assumptions, results of the analysis (i.e., stresses), and allowable limits specified in the ACI 349 Code has been provided in response to a previous NRC request for additional information. The SFP evaluation used no computer codes different from the codes of record. Refer to responses H.2 and H.3 in Reference 5.

J.13 *Does the licensee project modifications to piping or equipment supports for the proposed uprate? If any modifications are projected, provide examples of pipe supports requiring modification and discuss the nature of these modifications.*

J.13 Response

No modifications are required to piping or equipment supports as a result of the power uprate.

Additional Questions Regarding Previous NRC RAIs

Reference Question A.4

Verify equipment qualification under power uprate accident temperatures and pressures.

Response

The Byron Station and Braidwood Station EQ Program has demonstrated compliance with 10CFR50.49, "Equipment qualification of electric equipment important to safety for nuclear power plants." Documentation that demonstrates this qualification is on record at both stations.

For example, the ITT Barton Steam Generator Pressure Transmitters serve a two-fold function. First, their active safety function is to provide the main steam isolation signal or safety injection (SI) signal. The steam generator pressure transmitters must operate during the MSLB to transmit the low pressure signal that provides for closure of the MSIV within a few seconds after the pressure set point is reached. The transmitters were qualified, via type test, by the manufacturer to IEEE 323-74, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," requirements up to a peak temperature of 486°F. This temperature bounds the temperature at the time of MSIV closure for power uprate of 414°F, with significant margin.

The second function of the transmitters is to provide a R.G. 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," post accident monitoring indication. The manufacturer conducted an IEEE 323-74 test on the transmitter up to superheat/steam impingement conditions. The recorded temperature on the transmitter surface was 635°F. This temperature envelops the peak long term post accident temperature of 518°F, as determined by power uprate analysis, with significant margin.

The above summary is typical of the type of assessments performed for the 1E electrical equipment in the steam tunnel/valve rooms.

The test pressures used in the Byron and Braidwood Stations' EQ program bound the containment design pressure of 50 psig, and therefore bound the calculated peak pressures determined for the design basis accidents (i.e., LOCA, MSLB) under power uprate conditions. For pressure, qualification acceptability is determined by comparing the EQ program tested pressures against the calculated peak pressure. If the tested pressure value exceeds the calculated peak pressure value, with margin, qualification is acceptable. For EQ purposes, pressure effects are not time dependent. If the peak pressure has been addressed, so are the other lower pressures that can occur. This provides the rationale for not considering the pressure profile as a function of time.

Reference Question C.1

The ComEd heatup curves for Byron Station and Braidwood Station are less conservative than what was calculated by the staff by more than 10 °F at high pressure. An additional examination by the staff indicated that the heatup curve generated by P-T Calculator of EPRI agrees with the staff's values. To resolve this discrepancy, please provide the detailed calculation for the heatup curve (100 °F/hour) at high pressure (say 1500 psi) for Byron Station Unit 1.

Response

Exelon (formerly ComEd) used the Westinghouse methodology, also used at Kewaunee, Turkey Point, and North Anna Nuclear Stations. The Westinghouse methodology utilizes a finite element analysis to develop the thermal stresses. The results of the Westinghouse method are more accurate than would be generated using an "estimation" method, such as used by EPRI and the NRC. This more refined methodology results in lower thermal stresses, which ultimately allows for lower temperatures at higher pressures. This would explain the >10-degree difference noted. The following table shows temperatures and pressures for 100°F/hr heatup data for 22 and 32 EFPY curves for Byron Station Unit 1.

	Temp. of Water = 190°F	Temp. of Water = 195°F
Calculated Pressure	1466 psig (22 EFPY) 1406 psig (32 EFPY)	1566 psig (22 EFPY) 1499 psig (32 EFPY)
Wall Temp. Inner Rad.	189.46°F	194.465°F
Wall Temp. @ 1/4T	167.291°F	172.137°F
K_{IT} @ 1/4T	-13.9046 (KSI Sq. Rt. In)	-14.0264 (KSI Sq. Rt. In)
Wall Temp. @ 3/4T	142.883°F	147.541°F
K_{IT} @ 3/4T	10.5091(KSI Sq. Rt. In)	10.5982(KSI Sq. Rt. In)
Bending Stress @ 1/4T	-1.20419E+4 (PSI)	-1.21385E+4 (PSI)
Mean Stress @ 1/4T	3.10282E+3 (PSI)	3.12596E+3 (PSI)
Bending Stress @ 3/4T	1.38831E+3 (PSI)	1.4009E+3 (PSI)
Mean Stress @ 3/4T	2.81813E+3 (PSI)	2.84145E+3 (PSI)

Reference Question C.2

The ComEd response for Byron Station and Braidwood Station provides actual data to support the revised higher USE values for certain RPV welds. However, the reference was simply stated as "Record of Filler Wire Qualification Test." Please address differences, if any, in welding procedure and stress relief of the filler wire qualification test specimen and the actual RPV welds and assess their impact to the USE values.

Response

A description of all beltline materials data (i.e., chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data and initial RT_{NDT}) for Braidwood Station is contained in the documents indicated below.

Braidwood Station Unit 1

- Reference 1, Table 4-1, Heat Treatment History of the Braidwood Station Unit 1 Reactor Vessel Forgings and Weld Seam," provides the reactor vessel and surveillance material fabrication history.

- Reference 1, Table 4-5, "Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Station Unit 1 Weld Material Heat # 442011 (Using All Available Data)," and Table 4-6, " Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Station Unit 1 Forging Material 49D867-1/49C813-1," provide the current chemical analysis for the surveillance materials.

- Letter from R. M. Krich (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Reactor Pressure Vessel," dated September 3, 1998, provides the chemical analysis and the initial RT_{NDT} for the vessel beltline materials and demonstrates that the surveillance capsule materials are the limiting materials in the vessel for radiation embrittlement.

- WCAP-9807, "Commonwealth Edison Company Braidwood Station Unit No. 1 Reactor Vessel Radiation Surveillance Program," dated February 1981, provides the basis for the materials selection, unirradiated Charpy and tensile data for the surveillance materials, and drop-weight test results from the reactor vessel Certified Material Test Reports (CMTRs) for the limiting material.

Braidwood Station Unit 2

- Reference 2, Table 4-1, "Heat Treatment History of the Braidwood Station Unit 2 Reactor Vessel Forgings and Weld Seam," provides the reactor vessel and surveillance material fabrication history.

- Reference 2, Table 4-5, "Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Station Units 1 & 2 Weld Material Heat # 442011 (Using All Available Data)," and Table 4-6, " Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Station Unit 2 Forging Material 50D102-1/50C97-1," provide the current chemical analysis for the surveillance materials.

- Letter from R. M. Krich (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Reactor Pressure Vessel," dated September 3, 1998, provides the chemical analysis and the initial RT_{NDT} for the vessel beltline materials and demonstrates that the surveillance capsule materials are the limiting materials in the vessel for radiation embrittlement.

- WCAP-11188, "Commonwealth Edison Company Braidwood Station Unit No. 2 Reactor Vessel Radiation Surveillance Program," dated December 1986, provides the basis for the materials selection, unirradiated Charpy and tensile data for the surveillance materials, and drop-weight test results from the reactor vessel CMTRs for the limiting material.

Differences between the fabrication history of the surveillance material and that of the reactor vessel material for Braidwood Station are discussed below.

Braidwood Station Unit 1

Any differences between the fabrication history of the surveillance material and that of the reactor vessel material is contained in Table 4-1 in Reference 1. Specifically, the difference between the surveillance material and the vessel material is the heat treatment. The post weld stress relief of the surveillance material closely simulated that of the reactor vessel.

Braidwood Station Unit 2

Any differences between the fabrication history of the surveillance material and that of the reactor vessel material is contained in Table 4-1 in Reference 2. Specifically, the difference between the surveillance material and the vessel material is the heat treatment. The post weld stress relief of the surveillance material closely simulated that of the reactor vessel.

A description of all beltline materials data (i.e., chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data and initial RT_{NDT}) for Byron Station is contained in the documents indicated below.

Byron Station Unit 1

- The reactor vessel and surveillance material fabrication history is provided Table C.2-1 located at the end of this response.

- Reference 3, Table 4-5, "Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Station Unit 1 Weld Material Heat # 442002 (Using All Available Data)," and Table 4-6, " Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Station Unit 1 Forging Material 5P-5933," provide the current chemical analysis for the surveillance materials.

- Letter from William Levis (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Pressure Temperature Limit Report (PLTR)," dated July 12, 1999, provides the chemical analysis and the initial RT_{NDT} for the vessel beltline materials and demonstrates that the surveillance capsule materials are the limiting materials in the vessel for radiation embrittlement.

- WCAP-9517, "Commonwealth Edison Company Byron Station Unit No. 1 Reactor Vessel Radiation Surveillance Program," dated July 1979, provides the basis for the materials selection, unirradiated Charpy and tensile data for the surveillance materials, and drop-weight test results from the reactor vessel CMTRs for the limiting material.

Byron Station Unit 2

- Reference 4, Table 4-1, Heat Treatment History of the Byron Station Unit 2 Reactor Vessel Forgings and Weld Seam," provides the reactor vessel and surveillance material fabrication history.

- Reference 4, Table 4-5, "Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Station Units 1 & 2 Weld Material Heat # 442002 (Using All Available Data)," and Table 4-6, " Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Station Unit 2 Forging Material 49D330-1/49C298-1," provide the current chemical analysis for the surveillance materials.

- Letter from William Levis (Commonwealth Edison Company) to U. S. Nuclear Regulatory Commission, "Pressure Temperature Limit Report (PLTR)," dated July 12, 1999, provides the chemical analysis and the initial RT_{NDT} for the vessel beltline materials and demonstrates that the surveillance capsule materials are the limiting materials in the vessel for radiation embrittlement.

- WCAP-10398, "Commonwealth Edison Company Byron Station Unit No. 2 Reactor Vessel Radiation Surveillance Program," dated December 1983, provides the basis for the materials selection, unirradiated Charpy and tensile data for the surveillance materials, and drop-weight test results from the reactor vessel CMTRs for the limiting material.

Differences between the fabrication history of the surveillance material and that of the reactor vessel material for Byron Station are discussed below.

Byron Station Unit 1

Any differences between the fabrication history of the surveillance material and that of the reactor vessel material is shown in the table at the end of this response. Specifically, the difference between the surveillance material and the vessel material is the heat treatment. The post weld stress relief of the surveillance material closely simulated that of the vessel.

Byron Station Unit 2

Any differences between the fabrication history of the surveillance material and that of the reactor vessel material is contained in Table 4-1 in Reference 4. Specifically, the difference between the surveillance material and the vessel material is the heat treatment. The post weld stress relief of the surveillance material closely simulated that of the vessel.

Table C.2-1

Heat Treatment History of the Byron Station Unit 1 Reactor Vessel Forgings and Weld Seam

Reactor Vessel Forgings and Weld Seam			
Material	Temperature (°F)	Time (hrs)	Coolant
Lower Shell Forging 5P-5951	Austenitizing: 1630 ± 10	3.5 [a]	Water Quenched
	Re-Austenitizing: 1570 ± 10	3.5 [a]	Water Quenched
	Tempered: 1250 ± 10	5 [a]	Air Cooled
	Stress Relief: 1125 ± 25	50 [a]	Furnace Cooled
Intermediate Shell Forging 5P-5933	Austenitizing 1640 ± 10	3.5 [a]	Water Quenched
	Re-Austenitizing: 1590 ± 10	3.5 [a]	Water Quenched
	Tempered: 1240 ± 10	5.5 [a]	Air Cooled
	Stress Relief: 1125 ± 25	50 [a]	Furnace Cooled
Intermediate to Lower Shell Girth Weld Seam	Stress Relief: 1125 ± 25	50 [c]	Furnace Cooled
Surveillance Program Test Material			
Surveillance Program Test Forging 5P-5933	1640 ± 10	3.5	Water Quenched
	1590 ± 10	3.5	Water Quenched
	1240 ± 10	5.5	Air Cooled
	1125 ± 25	12.3	Furnace Cooled
Surveillance Program Test Weld Heat 442002	1125 ± 25	12.3	Furnace Cooled

a. Data obtained from Ladish Co. Material Test Reports.
b. The stress relief treatment received by the surveillance test forging and weldment has been simulated.
c. B&W Filler Wire Qualification Test, Test No. WF336, July 31, 1973

References for C.2:

1. WCAP-15316, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Station Unit 1Reactor Vessel Radiation Surveillance Program," Revision 1, October 1999.

2. WCAP-15369, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Station Unit 2 Reactor Vessel Radiation Surveillance Program," Revision 0, March 2000.

3. WCAP-15123, "Analysis of Capsule W from Commonwealth Edison Company Byron Station Unit 1 Reactor Vessel Radiation Surveillance Program," Revision 1, January 1999.

4. WCAP-15176, "Analysis of Capsule X from Commonwealth Edison Company Byron Station Unit 2 Reactor Vessel Radiation Surveillance Program," Revision 0, March 1999.

Reference Question C.2 – Additional Clarifying Information

The licensee's response provides actual data to support the revised higher USE values for certain RPV welds. However, the reference was simply stated as "Record of Filler Wire Qualification Test." Confirm that the "Record of Filler Wire Qualification Test" is just a different name for "Weld QTR" that was used in a previous submittal (ComEd letter of 11/19/93). Otherwise, the licensee needs to address differences, if any, in welding procedure and stress relief of the filler wire qualification test specimen and the actual RPV welds and assess their impact to the USE values. In addition, please address the staff comments in the following table to justify the use of the revised initial USE values proposed for the power uprate. The pages referenced in the "Staff Source" column refer to the 11/19/93 submittal.

Unit	Weld ID And Heat	USE (RVID)	Staff Source	Comment	USE (Uprate)
Byron 1	WF336 (442011) (Note: Heat # should be 442002)	74	Page 13	To abandon your previous conservative approach, you need to demonstrate that the USE of 78 ft-lb from "Weld QTR" were from three specimens. Otherwise, you need to use the average value of 78 and 74 reported on Page 13.	77
Byron 2	WF447 (442002)	67	Page 15	The Document stated, "The source of the 80 ft-lb value is undetermined.	80
	WF562 (442011)	70	Page 15	The document cited WCAP-12685 for this information	80
Braid-wood 1	WF562 (442011)	70	Page 10	To abandon your previous conservative approach, you need to demonstrate that the USE of 78 ft-lb from "Weld OTR" were from three specimens. Otherwise, you need to se the average value of 78, 70 and 71, reported on Page 10.	80
Braid-wood 2	WF562 (442011)	70	Page 10	Same as above	80

Response

As specified in ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for light Water Cooled Nuclear Power Reactor Vessels, E 706", Section 4.18, upper shelf energy level is defined as "the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy".

Table C.2-2 provides the basis for the vessel weld metal USE values. This information was taken from the manufacturer's Record of Filler Wire Qualification Test also referred to as the "Weld QTR" which is an acronym for weld qualification test record. These values, although not as conservative as those previously utilized for the response to Generic Letter 92-01 are the measured values for the vessel weld metal as defined by ASTM E185-82.

Table C2-2
Charpy V-Notch (CVN) Test Data / USE Valves

Unit	Weld ID & Heat #	CVN (FT/LBS)	SHEAR (%)	Temp. (°F)	USE Value
Byron 1	WF336 (442002) Wire Folio # 464-221	72 79 81 (a)	200 200 200	100 100 100	77
Byron 2	WF447 (442002) Wire Folio # 564-221	81 83 77 (b)	100 100 100	150 150 150	80
Braidwood 1/2	WF562 (442011) Wire Folio # 564-201	82 81 76 (c)	100 100 100	250 250 250	80

a. B & W Co., Record of Filler Wire Qualification Test, Test No. WF-336, 5/6/74
b. B & W Co., Record of Filler Wire Qualification Test, Test No. WF-447, 8/8/74
c. B & W Co., Record of Filler Wire Qualification Test, Test No. WF-562, 5/2/78

Reference Question E.2

Clarify the reason for reducing the operator response time for the feedwater line break event.

Response

The assumed operator response time for the feedwater line break event has been reduced from 30 minutes to 20 minutes to be consistent with the steamline break outside containment event. This is acceptable since both events can lead to a faulted steam generator and the procedure steps used by the operators to isolate the faulted steam generator are the same for both events. It has been demonstrated that the operator response time of 20 minutes can be achieved for the steamline break event. Westinghouse typically assumes an operator response time of 30 minutes for the steamline break outside containment event. For Byron Station and Braidwood Station, the response time assumed is 20 minutes.

Reference Question E.5

Confirm that any modifications to the training simulator will be made in accordance with ANS/ANSI 3.5.

Response

Simulator modifications for power uprate will be made in accordance with American Nuclear Standards and American National Standards Institute (ANS/ANSI) 3.5, "Nuclear Power Plant Simulators for Use in Operator Training," Section 5.3, "Simulator Modifications" and Section 5.4, "Simulator Testing." For power uprate, the station will provide initial training to the licensed operators prior to the on-line uprate. The data that will be used for the initial simulator modification will be the information from applicable power uprate calculations. The operators will be informed that the initial training they are receiving is based on the project calculations. After power uprate, plant data will be collected and compared to the simulator data to ensure the simulator meets the performance criteria specified in ANS/ANSI 3.5 Section 4.

Reference Question G.9

Regarding the Spurious Safety Injection event, what will be the temperature of the water being passed by the pressurizer safeties and what is the length of time the safeties are expected to pass water. Also discuss what Electric Power Research Institute (EPRI) tests are applicable to the Byron and Braidwood Stations condition.

Response

Results of testing by the Electric Power Research Institute (EPRI) support the conclusion that the spurious SI event would not transition into a higher Condition event. Although they may not be leaktight, the Pressurizer Safety Valves (PSVs) would close after passing water, and the leakage from up to three leaking PSVs is bounded by one fully open PSV. The "Inadvertent Opening of a Pressurizer Safety or Relief Valve" is analyzed as a Condition II event in the Byron Station and Braidwood Station UFSAR, section 15.6.1.

Relief of subcooled water was part of the EPRI testing of the Crosby safety valves (Reference EPRI Report #NP-2770-LD Volume 1 and 6). Two water relief tests were performed at a water temperature as low as 635 °F (i.e., Test #926 with lowest temperature between 635 °F and 640 °F, and Test #931 with lowest temperatures near 640 °F) and another performed at a water temperature of approximately 530 °F (i.e., Test #932). The results of the tests at 635 °F – 640 °F show stable valve operation. Valve chatter was experienced during the testing at 530 °F, resulting in damage to the valve internals. However, as indicated in EPRI Report #NP-2770-LD Volume 1, page S-6, in all cases, the safety valve closed in response to system depressurization.

The lowest water temperature predicted for the expected duration (i.e., 20 minutes) of the Spurious SI transient at Byron Station and Braidwood Station is significantly higher (i.e., 590 °F) than the lowest temperature (i.e., 530 °F) for the EPRI tests. Consequently, although stable valve operation cannot be assured, any valve damage would be expected to be less than the damage experienced during the EPRI testing. In any case, the safety valve will close upon system depressurization.

More importantly, it can be concluded that the Spurious SI event does not progress into a higher Condition transient (i.e., LOCA, Condition III). All three PSVs may lift in response to the event, but they will close and the resulting leakage from up to three PSVs is bounded by flow through one fully open PSV.

The assumed duration of the event is 20 minutes from initial SI signal to the time when system pressure is restored to below the PSV lift setpoint. Normally, the PORVs will automatically open by means of the control system grade actuation circuit, preventing the RCS pressure from ever reaching the PSV lift setpoint. The inadvertent SI event is terminated by operator action. Analyses show that during this 20 minute time frame, a PSV will cycle a number of times (i.e., approximately 20) with a duration of 5-8 seconds per cycle. Only one PSV is required to mitigate the pressure transient. Even though the three PSVs are set to lift at the same pressure, from a statistical standpoint, one valve would lift earlier than the other two. This would ensure that no more than one valve is challenged at a time. For the power uprate, the minimum expected discharge water temperature is 590 °F (at t = 20 minutes).