Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

September 20, 1988

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

Subject: Byron Station Units 1 and 2
Response to NRC Bulletin 88-08 &
88-08: Supplement 1 & 2
Docket Nos. 50-454/455

Reference: (a) NRC Bulletin No. 88-08
Dated June 22, 1988
(b) NRC Bulletin No. 88-08, Supplement 1
Dated June 24, 1988
(c) NRC Bulletin No. 88-08, Supplement 2
Dated August 4, 1988

Dear Sir:

The above referenced bulletin and supplements requested that licensees review the reactor coolant system (RCS) to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses and (2) take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses.

Commonwealth Edison has completed its review pursuant to the request outlined in Bulletin 88-08 and its supplements for Byron Station Units 1 & 2. Additionally Edison has discussed with the staff the methodologies used by Westinghouse Electric Corporation to support Byron Station's proposed response. These discussions were held per teleconferences September 2 and 9, 1988.

Per the staff's request Commonwealth Edison is submitting Byron Station's proposed response to Bulletin 88-08 along with the Westinghouse Electric Corporation engineering evaluation used to prepare this response. This information is attached in enclosures 1 & 2.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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1/1

Please address any questions that you or your staff may have concerning this response to this office.

Respectfully,

Wayne E Morgan

W.E. Morgan
Nuclear Licensing Administrator

Attachments:
cc: A.B. Davis
Resident Inspector/Byron

rf
5140K

Subscribed and Sworn to
before me this 30th day
of September, 1988

Delia I. Mayo
Notary Public

ATTACHMENT 1

Byron Station Response To NRC Bulletin 88-08: "Thermal Stresses in Piping Connected to Reactor Coolant System"

NRC Requested Action #1

Review systems connected to the RCS to determine whether unisolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping. For those addressees who determine that there are no unisolable sections of piping that can be subjected to such stresses, no additional actions are requested except for the report required below.

Byron Station's Response

A review of Byron piping systems connected to the RCS was conducted by Westinghouse to determine whether unisolable sections of connected piping could be subjected to stresses from temperature stratification or oscillations induced by leaking valves. Susceptible sections of piping were identified in the single Auxiliary Spray line and the four Charging Pump to Cold Leg Injection lines of each Byron Unit. Temperature oscillations may be induced in these lines by leakage of the isolation valves located between the RCS and the charging system.

NRC Requested Action #2

For any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses, examine nondestructively the welds, heat-affected zones and high stress locations, including geometric discontinuities, in that piping to provide assurance that there are no existing flaws.

Byron Station's Response

Current nondestructive examination technology does not permit reliable volumetric (radiographic/ultrasonic) examination of small diameter (less than 4-inch) stainless steel piping with high contact radiation readings. Therefore, the 1.5-inch diameter Charging Pump to Cold Leg Injection lines cannot be nondestructively examined at Byron Station. However, reasonable assurance can be provided that the thermal stress phenomenon potentially caused by the leakage of the Charging Pump to Cold Leg Injection Isolation Valves (1/2SI8801A,B) has not occurred at Byron Station. Performance of "Reactor Coolant System Isolation Valve Leakage Surveillance Procedure" (1/2 BVS 4.6.2.2-1) determines back leakage from the Reactor Coolant System through the Charging/Safety Injection Check Valves (1/2SI8815, 1/2SI8900A,B,C,D) to a test tap located between 1/2SI8815 and 1/2SI8801A,B valves. By virtue of the test tap's location, the detection of forward leakage from the Charging Pump through the 1/2SI8801A,B isolation valves is an unintended result of 1/2 BVS 4.6.2.2-1 performance. Procedures 1/2 BVS 4.6.2.2-1 are routinely performed as follows for the 1/2SI8815 check valves:

a) At least once per 18 months;
b) Prior to entering Mode 2 (Startup) whenever the plant has been in Cold Shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months;
c) Prior to returning valves 1/2SI8815 to service following maintenance, repair or replacement work on the valves;
d) Within 24 hours following 1/2SI8815 valve actuation due to automatic or normal action or flow through the valve.

Data from past performances of 1/2 BVS 4.6.2.2-1 were reviewed and indicate zero leakage in eleven performances and one instance in which leak rate was recorded as 0.000317 gallons per minute (negligible and well within the acceptance criteria of 1.0 gpm). Since all twelve leak tests of the 1/2SI8815 valves conducted from January 1985 to April 1988 indicate that zero or negligible leakage existed, 1/2SI8801A,B valves have not leaked; therefore, the Charging Pump to Cold Leg Injection lines have not been subjected to excessive thermal stresses.

Regarding the Auxiliary Spray line, Westinghouse recommended inspections of a 2-inch sockolet weld at the RCS piping connection and a portion of the 6-inch main spray piping. Due to nondestructive examination technique limitations, the 2-inch sockolet weld cannot be volumetrically examined, however, it will receive a surface examination and the susceptible portion of the 6-inch main spray line will be volumetrically examined prior to the end of the Unit 1 second refueling outage (scheduled to begin September 2, 1988). The necessity to nondestructively examine the Unit 2 main spray line and 2-inch sockolet weld will be determined by a pending stress analysis of the pipe. In the event that nondestructive examinations of the Unit 2 main spray line and 2-inch sockolet weld are required, the examinations will be completed prior to the end of the Unit 2 first refueling outage (scheduled to begin January 1989).

NRC Requested Action #3:

Plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit. This assurance may be provided by (1) redesigning and modifying these sections of piping to withstand combined stresses caused by various loads including temporal and spatial distributions of temperature resulting from leakage across valve seats, (2) instrumenting this piping to detect adverse temperature distributions and establishing appropriate limits on temperature distributions, or (3) providing means for ensuring that pressure upstream from block valves which might leak is monitored and does not exceed RCS pressure.

Byron Station Response:

In order to assure that the four Charging Pump to Cold Leg Injection lines on each Byron Unit will not be subjected to cyclic thermal stresses that could cause fatigue failure during the remaining lives of Byron Units 1 and 2, surveillance procedures will be developed or revised as necessary to require periodic tests specifically for leakage past the 1/2SI8801A,B isolation valves. If the 1/2SI8801A,B valves leak, the leakage will be discovered during the conduct of the surveillance, and action would ensue to determine the leakage source and correct the cause. Leak testing for the 1/2SI8801A,B valves will be routinely performed as follows:

a) At least once per 18 months;
b) Prior to entering Mode 2 whenever the plant has been in Cold Shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months;
c) Prior to returning valves 1/2SI8801A,B to service following maintenance, repair or replacement work on the valve;
d) Within 24 hours following 1/2SI8801A,B valve actuation due to automatic or manual action or flow through the valve.

The periodic performance of the leak test minimizes the potential for long term thermal cycling of the four unisolable Charging Pump to Cold Leg Injection lines by detecting and resolving isolation valve leakage. The 1/2SI8801A,B leak test procedures will be approved for use prior to the end of the Unit 1 second refueling outage (scheduled to begin September 2, 1988).

In order to assure that the single Auxiliary Spray line on Byron Unit 1 will not be subjected to cyclic thermal stresses that could cause fatigue failure, appropriate sections of the piping will be instrumented with external temperature monitoring devices. The outputs of these devices will be evaluated to determine if leakage past the single isolation valve from the charging system is occurring. The temperature monitoring devices will be installed on Unit 1 prior to the end of the second refueling outage (scheduled to begin September 2, 1988). If a pending stress analysis of the line concludes that the Auxiliary Spray line is not susceptible to fatigue failure, then the temperature monitoring devices may be removed. The necessity to instrument the Unit 2 Auxiliary Spray line will be determined by a pending stress analysis of the pipe. In the event that temperature monitoring of Unit 2 is required, the instrumentation will be installed prior to the end of the Unit 2 first refueling outage (scheduled to begin January 1989).

All Class 1 piping receives a VT-2 Visual Leak Test before and after refueling outages in accordance with the ASME Code. The examinations are intended to detect leakage from Class 1 piping. The Auxiliary Spray and Charging Pump to Cold Leg Injection lines are examined as part of the test, and any leakage due to thermal stress induced pipe cracking would be noted and resolved. Additionally, the high stress welds identified in these lines are routinely dye penetrant inspected as part of the ASME Inservice Inspection Program.

ATTACHMENT 2

Westinghouse Electric Corporation
Engineering Evaluation for
Commonwealth Edison
Byron and Braidwood Nuclear Stations
To Address NRC Bulletin 88-08

5140K

RAG

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

CAE-88-308
CCE-88-427
FSSE/SS-CAE-5530
S.O. CAE-280

August 15, 1988

Mr. D. Elias, Engineering Superintendent
Byron and Braidwood Stations
Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Commonwealth Edison Company
Byron and Braidwood Nuclear Stations
Revised Information Regarding Potential for
Temperature Oscillations in the Reactor Coolant Piping

Dear Mr. Elias:

The reference letter CAE-88-301 provided Westinghouse's initial input to Commonwealth Edison to address NRC Bulletin 88-08. This letter supersedes CAE-88-301 and provides more information about the screening criteria used to select potential locations of cyclic fatigue. The potential locations are listed on the attached table.

In the table, only those lines connected to reactor coolant system where temperature induced cyclic fatigue could occur, are included for both units 1 and 2 of Byron and Braidwood Stations. The screening criteria for determining these lines are as follows:

- Adequate driving force must be available. Only lines connecting the charging pumps to the RCS are potential candidates for continuous or cyclic leakage.

- Only lines with single, normally-closed isolation valves are included.

- 10CFR50.55a defines the reactor coolant pressure boundary as extending out to the second isolation boundary. Conservatively this would extend the piping line segments reportable under NRCB 88-08 Action 1 back to the second isolation valve; however, the location of the highest fatigue is the part of the system interconnecting piping between the last check valve and the reactor coolant main loop. The affected table contains only the portion of the piping between the last check valve and the main coolant loop.

- Piping peak-to-peak temperature cycles of 50°F were observed in the region of the cracked pipe at the J. M. Farley plant with the overall top-to-bottom stratification temperature being about 250°F. Westinghouse's recommendation to Portland Gas and Electric was that fatigue could occur at cycles as low as 20°F. All temperatures refer to piping outer diameter measurements.

- Piping layouts have been reviewed to locate cold traps. When piping runs vertically downward from heat sources (e.g., reactor coolant pipes or regenerative heat exchanger outlet piping), the water (and piping) at the bottom of the cold trap is cooler than the water and piping at the heat sources. This leads to the potential for temperature oscillations.

The charging lines and auxiliary spray line are potentially subject to temperature-iduced cyclic fatigue due to the piping layouts that have cold traps. Warm water (470°F) from the regenerative heat exchanger can propagate to the cold trapped piping where it will be further cooled. Further propagations of the now-cooled leakage to the pressurizer spray or cold leg(s) (530°F or 556°F, respectively) can cause cold-to-hot temperature cycles.

One of the charging isolation valves (CV8146 or CV8147) is closed at all times. This reduces the number of design transients that affect the charging nozzle because the operating duty is shared by each charging nozzle. At any given time, the charging line with the closed isolation valve could potentially have cyclic leakage and piping temperature oscillations. Valves CV8383A and B are spring loaded check valves with a cracking pressure of 200 psid intended to be thermal expansion relief valves for the regenerative heat exchanger tube side piping. The high cracking pressure prevents these valves from opening during normal operation, and the flow paths through valves CV8383A and B will not eliminate the potential for cyclic fatigue.

The charging/SI branch line potential leakage will be at the temperature of the charging pump discharge. Leakage could cause cold-to-hot temperature cycles to occur in the branch line piping adjacent to the reactor coolant cold legs.

The attached tables also define whether the potential leakage source temperature applies to normal plant operation or whether it is a transient condition. If it is a transient condition, a duration for the transient was defined. This information is provided to assist in the evaluation of the severity of any thermal stresses or to make the judgment that short-term transients will not cause high cyclic fatigue.

A separate document will be provided that documents the components and locations for non-destructive examination.

If we can be of any additional assistance, please call me.

Yours very truly,

S. A. Pujatas for
H. C. Walls, Manager
Commonwealth Edison Projects
U.S. Nuclear Projects

Attachment

cc: C. A. Moerke
R. A. Gesior
R. Pleniewicz
R. E. Querio
P. Thomas, W
R. W. Buchholz, W

ATTACHMENT

Line Number Subject to Temperature Stratification	Normally Closed Isolation Valves	Leakage Source Temperature (a) normal operation (b) design basis (c) design transient	Notes
RC37A3	CV8146	470°F(a)	1
		400°F(b)	1
		375°F(c)	2
		35°F(c)	3
RC28A3	CV8147	470°F(a)	1
		400°F(b)	1
		375°F(c)	2
		35°F(c)	3
RC30AA1-1/2	SI8801A or B	100-130°F(a)	1
	(valves in parallel)	60°F(b)	1
		35°F(c)	2
RC30AB1-1/2	SI8801A or B	100-130°F(a)	1
	(valves in parallel)	60°F(b)	1
		35°F(c)	2
RC30AC1-1/2	SI8801A or B	100-130°F(a)	1
	(valves are in	60°F(b)	1
	parallel)	35°F(c)	2

ATTACHMENT

Line Number Subject to Temperature Stratification	Normally Closed Isolation Valves	Leakage Source Temperature (a) normal operation (b) design basis (c) design transient	Notes
RC30AD1-1/2	SI8801A or B (valves are in parallel)	100-130^{o}F(a)	1
		60^{o}F(b)	1
		35^{o}F(c)	2
RY18A2	CV8145	470^{o}F(a)	1
		400^{o}F(b)	1
		375^{o}F(c)	2
		35^{o}F(c)	3

Notes

1) Potentially, leakage source would occur continuously.

2) Leakage source temperature would occur very infrequently--likely less than one hour per year per reactor.

3) Leakage source temperature based on low pressurizer level signal without coincident safety injection or on phase A containment isolation without coincident safety injection. These scenarios are unlikely to continue for more than 0.5 hours per year (for one instance per reactor) until either operator intervention/rectification occurs or else the scenario degrades to a safety injection which isolates normal charging.