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cc-906D

December 4, 1980
G02-80-279

Docket No. 50-397

Mr. R. L. Tedesco, Assistant Director, Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Tedesco:

Subject: CRACKING OF BWR JET PUMP HOLDDOWN BEAMS

Ref.: Letter, R. L. Tedesco to N. O. Strand, same subject, dated August 5, 1980

FILE COPY

The referenced letter requested the Supply System provide specific information regarding actions taken to preclude cracking of jet pump holddown beams. The following responses correspond directly with the questions posed in your letter.

1. WNP-2 jet pump beams have been installed, but will be retensioned from a 30 kip preload to a 25 kip preload before fuel load. This is expected to increase beam operating time to crack initiation at the 2.5% probability level to a range of 19 to 40 years

   During operation, periodic inspections will be conducted as part of our overall inservice inspection program. Inspection frequencies will be developed in the future based on lead plant inspection results and the results of future GE testing. These inspections should provide adequate warning of potential beam failure.

2. It is our position that reducing the tension preload to 25 kips on the beams provides an adequate long term solution. If a problem is still present, as identified by our inservice inspections, improved heat treated beams may be purchased from GE. Tests indicate the improved beams may provide double the time to crack initiation as compared to the current beams.

no letterhead rec'd

AUTHOR	KA Hadley [signature]		FOR SIGNATURE OF:	GD Bouchey	
SECTION				[signature]	
FOR APPROVAL OF	GC Sorensen	LT Harrold	DL Renberger		
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3. This item is answered by both above items 1 and 2.

Very truly yours,

Original Signed By:

G. D. Bouchey
Director, Nuclear Safety

bk

cc: N. S. Reynolds, D&L
W. Woods, NUS
M. C. Lynch, US NRC
B. J. Youngblooc, US NRC

AUTHOR			FOR SIGNATURE OF		
SECTION					
FOR APPROVAL OF					
APPROVED					
DATE					

WP 114

QUESTION NO. 48

Provide a commitment to NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking".

RESPONSE

The Supply System's response to NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Line Nozzle Cracking", will be completed by January 8, 1982. The current status of our position on the feedwater and CRD cracking problems is as follows:

Feedwater Nozzle

- The WNP-2 welded sparger feedwater nozzle is an NRC approved design (NUREG-0619, Page 15).
- The WNP-2 nozzles are unclad.
- The RWCU System has been rerouted so it discharges to all nozzles.
- The need for a low flow feedwater controller as described in GE report NEDE 21821-A has not yet been established. WNP-2 current design employs a low flow feedwater controller. The Supply System is currently evaluating whether or not the existing controller meets the intent of NUREG-0619 and NRC generic letter 81-11.
- An augmented Inservice Inspection program for the feedwater nozzle has been submitted in response to FSAR Question 121.8. This response will be considered in our reply to NUREG-0619.

CRD Return Line

- CRD return line has been cut and capped as allowed by NUREG-0619, Page 31.
- CRD return line has been rerouted through redundant equalizing valves to the exhaust water header.
- The control rod drive preoperational test will demonstrate that the system is fully operational and that all components including the hydraulic drive mechanisms, pumps, and flow control valves function properly. The CRD System will be configured with the modifications noted in the NRC concern.

- o In order to assure satisfactory system operation with the single failure of an equalizing valve, the proposed design modification will include the addition of two equalizing valves installed in a parallel configuration. The failure of either valve will not impair CRD operation for any foreseen operating or accident condition.

- o There will be no increased potential for carbon steel corrosion products to be deposited in the drives. All lines in the WNP-2 Hydraulic System after the drive water filters are made of stainless steel.

- o The NRC requested GE by letter of January 28, 1980, to recalculate the makeup flow capacity for the 251-inch BWR-5 without the CRD return line. This generic information has been provided by letter of May 2, 1980, from Mr. R. L. Gridley, GE to Mr. D. G. Eisenhut, NRC, concurrently with this docketed response for LaSalle. The results indicate that the 251-inch BWR-5 CRD system without a return line (capped Nozzle 10) can achieve a vessel makeup flow in excess of its calculated boiloff rate of 180 gpm. This confirms the same boiloff rate as previously documented in a March 14, 1979, submittal from GE. Furthermore, since the CRD system is not designed to perform an ECCS function, the additional testing to demonstrate the required return-flow capacity to the vessel is not warranted.

Summation - This item is closed.

QUESTION NO. 49

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specification which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

RESPONSE

The valves which separate the Reactor Coolant System (RCS) from interfacing low pressure systems are listed in Table I.

These valves are included in the WNP-2 Pump and Valve Inservice Testing Program which was developed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV. The Supply System's position is that the requirements of the Code provide adequate assurance of valve integrity. Specifically:

A) The Supply System will leak rate test the valves listed in Table I at least every two years (IWV-3422). This position is justified by the following:

1. All the valves listed in Table I have direct monitoring position indication which verifies valve position in the Control Room.

2. The low pressure portions of these interfacing systems are protected against an intersystem LOCA by the following:

a) The normal functional differential pressure forces the check valves on their seats. The air operator of these testable check valves cannot open the valves at normal differential pressure. (HPCS-V-5, LPCS-V-6, RHR-V-41A, B, C, RHR-V-50A, B, RCIC-V-66).

b) Electrical interlocks prevent the motor-operated valves from opening when the differential pressure across the valve exceeds specified limits (LPCS-V-5, RHR-V-42A, B, C) or when the RCS pressure exceeds specific values (RHR-V-53A, B, RHR-V-8, RHR-V-9, RHR-V-23, RHR-V-123A, B).

c) Whenever excessive leakage is present at a pressure boundary isolation valve, this leakage will increase pressure in the downstream side of these systems which will annunciate a high pressure alarm.

d) Excessive leakage will be channeled into the suppression pool where an increase in suppression pool level will be indicated.

e) The high pressure core spray pump suction piping is protected by an additional check valve on the pump discharge.

B) The Supply System will specify the leak test medium and the test acceptance criteria as permitted by the ASME Code (IWV-3425 & 3426).

C) The periodic leak test will be done prior to entering Operational Condition 2.

D) After maintenance which is deemed by the Owner to affect leak tightness of the valve, leak testing will be performed in accordance with ASME Section XI prior to the valve's returning to service.

Summation - The NRC needs to have their IST people meet with the Supply System. NRC will set up a meeting and get back to us.

TABLE I

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Valve Number	Valve Type	Size		Function
HPCS-V-4	G	12"	(O)	HPCS injection line containment isolation valve
HPCS-V-5	TC	12"	(i)	
LPCS-V-5	G	12"	(o)	LPCS injection line containment isolation valve
LPCS-V-6	TC	12"	(i)	
RCIC-V-66	TC	6"	(i)	RCIC Rx head spray containment isolation valve
RCIC-V-13	G	6"	(o)	
RHR-V-8	G	20"	(i)	RHR shutdown cooling supply to RHR pumps containment isolation valves
RHR-V-9	G	20"	(o)	
RHR-V-23	Globe	6"	(o)	RHR to Rx head spray containment isolation valve
R-V-41A, B, C	TC	14"	(i)	RHR LPCI containment isolation valve
RHR-V-42A, B, C	G	14"	(o)	
RHR-V-50A, B	TC	18"	(i)	RHR shutdown cooling return line to Rx vessel containment isolation valve
RHR-V-53A, B	Globe	18"	(o)	
RHR-V-123A, B	G	1"	(i)	By-pass valve for V-50A, B containment isolation valve

o - outside
i - inside
G - gate
TC testable check

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LaSalle Tech Spec.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24 hour period.

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d. 1 gpm leakage at a reactor coolant system pressure at {1000} ± {10} psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

delete

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c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric particulate and gaseous radioactivity at least once per 12 hours,

b. Monitoring the primary containment sump flow rate at least once per 12 hours, and

c. Monitoring the primary containment air coolers condensate flow rate at least once per 12 hours.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months. prior to entering operational condition 2

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b. Prior to entering 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 the valve to service following maintenance, repair or replacement work on the valve.

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d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITIONS 2 or 3.

WNP-2 DSER

QUESTION NO. 50

Does the design criteria for component supports in WNP-2 systems categorize the stresses produced by seismic anchor point motion of piping and the thermal expansion of piping as primary or secondary?

NRC position - For the design of supports, these stresses should be considered primary. Expansion stresses in the support themselves may be categorized as secondary.

RESPONSE

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement (thermal & seismic anchor point) are treated as primary or secondary stresses, per ASME Section III, for piping. For loads due to relative displacement, supports are analyzed as per ASME Code Section NF. The applicants are committed to assess the design with respect to the NRC position.

10/16/81

Item 1 WNP-2 will provide information to indicate those areas on the RPV that can't be fully mechanically inspected by inservice examinations.

Item 2 WNP-2 will provide an update of the list of non-examinable welds (provided in response to 121.19) in January 1982.

Item 3 (Q 121.20) WNP-2 will provide an expanded response to discuss the design vs. operating temperature and pressure on those lines exempted from Preservice volumetric and/or surface examination based on paragraph IWC-1220(a) of section XI for those lines to which this exemption criteria was applied that have a design temperature of 212^o (versus 200^o).

Item 4 (Q 121.20) WNP-2 will revise the response to indicate that a volumetric examination in lieu of a surface examination for HPCS piping $>$1/2" thick will be conducted on 10% of those welds exempted by chemistry; otherwise the commitment of 10% sample by a surface method was acceptable.

Additionally a table itemizing those welds exempted by chemistry will be attached to the response of Q 121.20, showing sizes, and operating temperatures and pressures.

Item 5 (Q 121.21) The response to this question will be revised to add the following paragraph:

> "During the conduct of inservice examinations the criteria for evaluating a crack-like indication will not be limited to signal amplification alone. Appropriate consideration will be given to other factors such as the location of the indication. Those indications determined to be crack-like will be evaluated."

Item 6 (Q 121.22) WNP-2 will revise the response to this question to add the following statements:

"Any additional welds which require PSI will be included in that program in Amendment No. 3 scheduled for submittal 12/81.

Figures 3.6-147a through d in section 3.6 of the FSAR define the break exclusion areas. Section 3.6.1 commits to an augmented inservice inspection program on these lines. Section 5.2.4 of the FSAR will be amended to reflect these augmented requirements. (Section 5.2.4 draft attached.)

For those lines beyond the outboard containment isolation valves which are not normally pressurized, the inspection boundary will stop at the outboard isolation valve. This approach is consistent with that taken by the piping designer. No pipe whip restraints are installed beyond the containment isolation valve by the designer because these lines are not pressurized and therefore are not subject to pipe whips."

5.2.4.9 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

An augmented Inservice Inspection Program will be implemented for WNP-2, on highenergy* Class 1 piping systems which penetrate containment for which the effects of postulated pipe breaks would be unacceptable. This program will entail a volumetric examination of all circumferential butt welds (surface examination for socket welds) between the first pipe whip restraint beyond the inside containment isolation valve, and first pipe whip restraint beyond the outside containment isolation valve on high-energy Class 1 lines greater than one (1) inch which penetrate the containment.

In those cases where the piping beyond the containment isolation valve is not pressurized (i.e. low energy), the augmented Inservice Inspection boundary will stop at the containment isolation valve.

This program will include branch lines which fall within the augmented Inservice Inspection boundary to the first pipe whip restraint beyond the branch line isolation valve on the first normally closed valve, whichever comes first.

*High-energy lines include those systems that, during normal plant conditions, are either in operation or maintained pressurized and where either the maximum operating pressure exceeds 275 psig or maximum operating temperature exceeds 200°F. If, for a particular line, the above pressure and temperature limits are not exceeded more than 2% of the time that the system is in operation, then that line is considered moderate energy and is exempt from the requirement for augmented Inservice Inspection.

10/16/81

Item 7

WNP-2 will submit a relief request for the requirement to perform a surface vs. a volumetric examination on the RHR pump casing welds. (Question 3)

Item 8

Include a statement in the executive summary to the PSI program that 1/2 inch wall thickness, which is part of the Summer 1978 Code, has been used in this program. (Question 4)

Item 9 WNP-2 will include a statement in the executive summary of the PSI program that the four inch nominal pipe size is taken from the 1978 Summer Addenda of ASME section XI.

Item 10

WNP-2 will provide a statement in the executive summary of the PSI program plan stating that bolting examinations in categories B-G-1, B-G-2 and C-D to ASME section XI 1977 edition to Summer 1978 Addenda will be accomplished.

Item 11 WNP-2 will add a statement to the executive summary of the PSI program plan to indicate that the Summer 1978 Code is used in performing branch connection examinations.

Item 12 WNP-2 will provide a statement to the executive summary of the PSI program to indicate that RPV nuts and studs were examined per the Summer 1978 Addenda to section XI.

Item 13 WNP-2 will submit a relief request for those RPV and piping welds where full code compliance cannot be accomplished. (Submit by 12/14/81.)

10/16/81

Item 14 WNP-2 will provide a description of the Reactor vessel examination made in the PSI program executive summary.