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November 18, 1988

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

SUBJECT: Leak-Before-Break (LBB) submittal (TAC 68318)

Attached is our response to the request for additional information transmitted in the course of discussion with the NRC staff on October 27, 1988.

If you have any further questions, please contact Mr Jude G. Del Percio, Manager, Regulatory Affairs.

Very truly yours,

Steph Bram

Attachments

cc: Mr. William Russell
Regional Administrator - Region I
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Attachment

Response to Request for Additional Information on
Leak-Before-Break Submittal
Dated May 23, 1988

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
November 18, 1988

Question 1

Why is "Leak-Before-Break" needed for OFA submittal?

Response

The LOCA analysis for OFA reload performed recently by Westinghouse assumes that the leak-before-break methodology submitted on May 23, 1988 would be approved for the primary reactor coolant loop. As such, the most limiting branch line piping (the accumulator line) was assumed to rupture as a basis for the reload/OFA fuel LOCA forces analysis. The accumulator line break is modeled in the same way as a limited displacement break of the reactor coolant inlet nozzle, except that a smaller break area located away from the vessel inlet is presumed. The reduced break area as well as the additional distance between the vessel inlet and the break account for the reduction in the magnitude of the loads. This break yielded fuel grid loads at Indian Point Unit 2 (IP-2) which were well below (50%) the allowable limit, and as such no grid deformation was predicted to occur. As a result, the associated reload/OFA fuel ECCS analysis for IP-2 did not have to account for effects of deformed grids.

Question 2

Is Indian Point 3 a twin to Indian Point 2 for the purposes of Leak-Before-Break?

Response

We do not know. With respect to the design of the NSSS, the reactor core and internals, Indian Point 3 (IP-3) is technically equivalent to IP2. Our understanding is that IP-3 has LBB technology approved without any changes to the NSSS design. As a design twin to IP-3, LBB technology applied to IP-2 does not introduce any technical difference.

Question 3

Provide a list of plants that have LBB technology approved.

Response

Plant	Approval/SER date
Seabrook	1/18/85
Catawba	4/23/85
Vogtle	6/13/85
Cook	11/22/85
Bellafonte	2/18/86
Indian Point 3	3/10/86
Point Beach	5/08/86
Mcguire	5/08/86
Comanche Peak	6/08/87
Ginna	9/09/86
Wolf Creek	10/28/86
Praire Island	12/22/86
Yankee Rowe	8/04/87
WNP 1	9/02/87
Beaver Valley 1	10/21/87
TMI 1	11/05/87
Trojan	8/05/88

Question 4

What are the impacts of removing the assumption that LBB technology has been approved from the OFA submittal?

Response

The OFA reload license amendment application would need to be resubmitted with additional analyses considering LOCA hydraulic forces induced by breaks of the primary reactor coolant loop in the calculation of fuel assembly structural integrity. This would involve significant time and cost allocation. Westinghouse estimates that such a reanalysis could take from 3 1/2 to 5 1/2 months to complete prior to resubmittal to NRC. This would significantly compress review schedules to meet our March, 1989 refueling outage needs.

Question 5

What are the other benefits of LBB?

Response

IP-2's Steam Generator snubber reduction program will also utilize LBB technology. Reduction in the number of large bore snubbers supporting the IP-2 Steam Generators will reduce plant maintenance and surveillance cost and enhance the reliability of primary loop support system. In addition, elimination of pipe whip restraints and jet shields would facilitate in-service inspection and routine maintenance. Removal of pipe whip restraints inside containment would eliminate personnel radiation exposure presently necessitated by restraint inspection, including gap clearance inspection, and disassembly for pipe weld inspections. Removal of this equipment would also provide greater access to piping and other equipment for maintenance activities. Inspection savings would also alleviate personnel and equipment costs.

Question 6

What is the historical background of LBB?

Response

In 1975 to 1978 the NRC required licensees to apply blowdown loads at adverse locations in the RCS loop piping, resulting in asymmetric blowdown effects on the RCS components (such as reactor vessel, pumps, steam generators) as well as the core. Of particular concern to the NRC during this time was maintaining a coolable core geometry after application of the asymmetric blowdown loads, and supporting the RCS components so that they remained integral (except for the broken loop where the postulated LOCA occurred). Initially, the NRC was most concerned with a break occurring inside the reactor cavity where the break reaction forces could potentially disturb the core and vessel geometry sufficiently to challenge adequate core cooling. Some plants, including IP-2, installed primary shield wall restraints (wagon-wheel) to prevent full area double ended blowdown from occurring, thereby substantially reducing the reaction forces on the core and vessel. During 1978 to 1980, it became clear to the NRC and licensees that if hypothetical worst case asymmetric blowdown loads were presumed for analytical purposes, then extensive plant modifications for all plants would be required. An owners group was formed to address this issue generically. The Owners Group applied state of the art elastic-plastic fracture mechanics technology and concluded that given a flaw within the detection capability of current inservice examination techniques, no instantaneous pipe rupture could occur since the flaw would first propagate into a leak that would stablize and be detected by ordinary leakage detection systems.

The results of the analysis shows that leak-before-break will occur. In 1984, the NRC issued a Safety Evaluation Report (Generic Letter 84-04) accepting leak-before-break technology. This action eventually led to the May, 1986 final rule amending GDC-4.