CENTRAL FILES

GL-80-12

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

February 8, 1980

Docket Nos. 50-03
50-247

Consolidated Edison Company of
New York, Inc.
ATTN: Mr. W. J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

The enclosed IE Bulletin No. 80-04, is forwarded for action. A written response is required. If you desire additional information regarding this matter, please contact this office.

Sincerely,

Robert T. Carlson
for Boyce H. Grier
Director

Enclosures:
1. IE Bulletin No. 80-04
2. List of Recently Issued IE Bulletins

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(215-337-5253)

cc w/encls:
L. O. Brooks, Project Manager, IP Nuclear
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ENCLOSURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

SSINS No.: 6820
Accessions No.:
7910250504

ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

Description of Circumstances:

Virginia Electric and Power Co. submitted a report to the Nuclear Regulatory Commission dated September 7, 1979 that identified a deficiency in the original analysis of containment pressurization as a result of reanalysis of steam line break for North Anna Power Station, Units 3 and 4.

Stone and Webster Engineering Corporation performed a reanalysis of containment pressure following a main steam line break and determined that, if the auxiliary feedwater system continued to supply feedwater at runout conditions to the steam generator that had experienced the steam line break, containment design pressure would be exceeded in approximately 10 minutes. The long term blowdown of the water supplied under runout conditions by the auxiliary feedwater system had not been considered in the earlier analysis.

On October 1, 1979, the foregoing information was provided to all holders of operating licenses and construction permits in IE Information Notice No. 79-24. The Palisades facility did an accident analysis review pursuant to the information in the notice and discovered that with offsite power available, the condensate pumps would feed the affected generator at an excessive rate. This excessive feed was not considered in the analysis for the steam line break accident.

On January 30, 1980, Maine Yankee Atomic Power Company informed the NRC of an error in the main steam line break analysis for the Maine Yankee plant. During a review of the main steam line break analysis, for zero or low power at the end of core life, the licensee identified an incorrect postulation that the startup feedwater control valves would remain positioned "as is" during the transient. In reality, the startup feedwater control valves will ramp to 80% full open due to an override signal resulting from the low steam generator pressure reactor trip signal. Reanalysis of the event shows the opening of the startup valve and associated high feedwater addition to the affected steam generator would cause a rapid reactor cooldown and resultant return-to-power, a condition outside the plant design basis.

Actions to be Taken by the Licensee:

For all pressurized water power reactors with an operating license and those reactors listed in Attachment 1:

1. Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

2. Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

   a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

   b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

   c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

   d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

3. If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed.

4. Within 90 days of the date of this Bulletin, complete the review and evaluation required by this Bulletin and provide a written response describing your reviews and actions taken in response to each item.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

Enclosure 1

For boiling water reactors with an operating license or a construction permit and all pressurized water reactors with a construction permit, not listed in Attachment 1, this Bulletin is for information purposes only and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Plants with construction permits that are required to respond to the bulletin:

Diablo Canyon
McGuire
North Anna 2
Salem 2
Sequoyah

If the permit holders have responded to earlier requests from the NRC on some of the items presented in the bulletin, they may respond to the bulletin by reference to the response to the earlier request.

ENCLOSURE 2

RECENTLY ISSUED IE BULLETINS

Bulletin No.	Subject	Date Issued	Issued To
79-25	Failures of Westinghouse BFD Relays in Safety-Related Systems	11/2/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) (for Action)
79-02 (Rev. 2)	Pipe Base Plate Designs Using Concrete Expansion Bolts	11/8/79	All Power Reactor Facilities with an OL or CP
79-26	Boron Loss From BWR Control Blades	11/20/79	All BWR Power Reactor Facilities with an OL
79-27	Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation	11/30/79	All Power Reactor Facilities with an OL and those nearing Licensing (for Action) All Power Reactor Facilities with a CP (for Information).
79-28	Possible Malfunction of NAMCO Model EA180 Limit Switches at Elevated Temperatures	12/7/79	All Power Reactor Facilities with an OL or CP
79-01B	Environmental Qualification of Class IE Equipment	1/14/80	All Power Reactors with an OL except SEP Plants
80-01	Operability of ADS Valve Pneumatic Supply	1/14/80	All BWRs with an OL
80-02	Inadequate Quality Assurance for Nuclear Supplied Equipment	1/21/80	All BWRs with an OL or CP
80-03	Loss of Charcoal From Standard Type II, 2 Inch, Tray Adsorber Cells	2/6/80	All Power Reactor Facilities with an OL or CP