June 10, 1981

Docket No. 50-219
LS05-81-06-028

RECEIVED
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U.S. NUCLEAR REGULATORY COMMISSION

Mr. I. R. Finfrock, Jr.
Vice President - Jersey Central
Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

SUBJECT: OYSTER CREEK - SEP TOPIC III-5.A, EFFECTS OF PIPE BREAK ON STRUCTURES, SYSTEMS AND COMPONENTS INSIDE CONTAINMENT

Enclosed is our draft evaluation of SEP Topic III-5.A. This assessment compares your facility with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 30 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject is modified before the integrated assessment is completed.

As noted in the enclosure, there are some areas in the interaction studies for which we require additional information. Please provide your schedule for completion within 30 days.

In future correspondence regarding this topic, please refer to the topic number in your cover letter.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

8106160531

Enclosure:
As stated

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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TABLE OF CONTENTS

SEP EVALUATION

OF

EFFECTS OF PIPE BREAK ON STRUCTURES,

SYSTEMS AND COMPONENTS INSIDE CONTAINMENT

TOPIC III-5.A

FOR THE

OYSTER CREEK NUCLEAR POWER PLANT

I. INTRODUCTION

The safety objective of Systematic Evaluation Program (SEP) Topic III-5.A, "Effects of Pipe Break on Structures, Systems and Components Inside Containment," is to assure that pipe breaks would not cause the loss of needed funtion of "safety-related" systems, structures and components and to assure that the plant can be safely shut down in the event of such breaks. The needed functions of "safety-related" systems are those functions required to mitigate the effects of the pipe break and safely shut down the reactor plant.

II. REVIEW CRITERIA

The current criteria for review of pipe breaks inside containment are contained in Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," including its attached Branch Technical Position, Mechanical Engineering Branch 3-1 (BTP MEB 3-1).

III. RELATED SAFETY TOPICS AND INTERFACES

1. This review complements that of SEP Topic VII-3, "Systems Required for Safe Shutdown."

2. The environmental effects of pressure, temperature, humidity and flooding due to postulated pipe breaks are evaluated under Unresolved Safety Issue A-24, "Qualification of Class IE Safety-Related Equipment."

3. The effects of potential missiles generated by fluid system ruptures and rotating machinery were also considered and are evaluated under SEP Topic III-4.c, "Internally Generated Missiles."

4. The effects of containment pressurization are under SEP Topic VI-2.D, "Mass and Energy Release for Possible Pipe Break Inside Containment."

5. The original plant design criteria in the areas of seismic input, analysis design criteria are evaluated under SEP Topic III-6, "Seismic Design Consideration."

IV. REVIEW GUIDELINES

The licensee's break location criteria and methods of analysis for evaluating postulated breaks in high energy piping systems inside containment have been compared with the currently accepted review criteria as described in Section II above. The review relied upon information submitted by the licensee, Jersey Central Power and Light Company (JCP&L), in References 1, 2 and 3.

When deviations from the review criteria are identified, engineering judgement is utilized to evaluate the consequence of postulated pipe break and to assure that pipe break would not cause the loss of needed function of "safety-related" systems, structures, and components and to assure that the plant can be safely shutdown in the event of such break.

V. EVALUATION

A. BACKGROUND

On July 20, 1978, the SEP Branch sent a letter (Reference 4) to KMC, Inc. requesting an analysis of the effects of postulated pipe breaks on structures, systems and components inside containment. In that letter, the staff included a position that stated three approaches were appropriate for postulating breaks in high energy piping systems (Either $p \geq 275$ psig or $T \geq 200°F$). The approaches are:

1. Mechanistic
2. Simplified Mechanistic
3. Effects Oriented

The staff further stated that combinations of the three approaches could be utilized if justified.

B. BREAK CRITERIA

In response to our letter, the licensee submitted Reference 1 which summarized all information contained in the Oyster Creek docket pertaining to SEP Topic III-5.A. Attachment 7.c of the document was originally submitted to the NRC on July 15, 1974, in Reference 5, in response to NRC Question 14.c. It included a report of pipe break evaluations entitled, "Analysis of Pipe Breaks Inside Containment." The licensee has identified postulated pipe break locations in high energy piping systems (either $p>275$ psig or $T>200°F$) using a mechanistic approach, i.e. a criteria based on stresses in the piping system. In summary, this work resulted in the identification of 150 postulated breaks in accordance with the criteria of Regulatory Guide 1.46. In general, this criteria postulated pipe breaks at terminal ends and at intermediate locations where the calculated stress exceeded a threshold level, i.e. $0.8\ (S_h + S_a)$, but at at least two intermediate locations. The loading conditions of gravity stress, pressure stress, seismic stress and thermal stress were considered and combined.

In evaluating criteria for postulated pipe break effects on structures, systems and components inside containment, we find it is acceptable to apply Regulatory Guide 1.46 break criteria for ASME Section III Code Class 2 and 3 piping systems to piping systems designed and analyzed in accordance with USAS B31.1.0. However, based on information provided in this review, it is not clear that the seismic stresses have been conservatively calculated. In some cases, the maximum seismic stress determined by static analysis is used at all locations in the system. A comparison of these seismic stresses with two available piping calculations (Feedwater piping and Main steam piping) from Reference 6, which is performed for SEP TOPIC III-6, indicates differences in points of high stress in the main steam piping system. This will result in different postulated pipe break locations. Nevertheless, it must be noted that the licensee has identified 150 postulated breaks which is about the same number that a new plant under licensing review would have for inside contaiment. Therefore, our assessment based on the information currently available is that the licensee has provided for a spectrum of postulated pipe breaks which include or envelope the most likely break locations inside the containment. We recommend that any major deviations identified in future studies from SEP TOPIC III-6 be incorporated into a final reassessment of Reference 2.

C. INTERACTION EVALUATIONS

The licensee has developed a simplified matrix to study the interaction consequences of these break locations on safety-related systems, structures and components. This interaction study grouped the 150 postulated breaks into three different categories as follows:

- Category 1 included 89 postulated breaks which were considered to have no potential adverse consequences and, therefore, no modifications to the plant are necessary.
- Category 2 included 48 postulated breaks which could have potential adverse consequences, but which occurred at locations of low stress, i.e. the stresses were less than 50% of the stress criterion of Regulatory Guide 1.46. The postulated breaks in this category were considered not to require specific failure protection because they occur at locations of low stress.
- Category 3 included 13 postulated breaks which could have potential adverse consequences and which occurred at locations of higher stress, i.e. the stresses were 50% or greater of the stress criterion of Regulatory Guide 1.46. An augumented inservice inspection program modified to be consistent with the NRC request (Reference 8) has been implemented for these locations (Reference 9) since 1975.

Potential adverse consequences (Category 2 and 3 breaks) were defined as any interaction with containment, with piping in other high energy lines or safety systems of smaller size, or with electrical and mechanical equipment having a safety function.

On February 6, 1979, JCP&L revised some of the seismic stress analysis. The effect of the revised seismic stress was to reduce the number of breaks from 150 to 141.

Based on discussions with the licensee during a meeting on February 6, 1979 and subsequent telephone conversations, NRC requested further information to clarify the consequences of postulated breaks in categories 2 and 3. In response to the NRC request, the licensee submitted Reference 3.

In this report, the licensee evaluated the plant's capability to reach safe shutdown with the dynamic effects of the postulated breaks. The general approach taken was to ensure that no break could affect more than one train of a safety system, and that the intact equipment still contained sufficient redundancy to withstand a single active failure without loss of safety function. Interaction with the containment wall was reexamined to determine if the breaks could penetrate the drywell wall. This method is a refinement of the original interaction matrix, taking into account the functions of affected systems, physical separation and redundancy. The licensee's conclusion from this evaluation was that all breaks had acceptable consequences.

For the Oyster Creek facility, the systems that would be used to reach safe shutdown following a high energy line break inside containment are the following:

Emergency Condensers
Core Spray Systems
Automatic Depressurization System (ADS)
Instrumentation (reactor water level and reactor pressure)
Motive and control power

These systems could also be used to achieve safe shutdown in the absence of a break as shown in Reference 10.

Many of the components in the above systems, such as the core spray pumps, and the emergency condensers themselves are located outside containment, and thus would not be affected by the break inside containment.

The two low pressure core spray piping systems are physically separated from each other within containment. In addition, each train of the core spray system can withstand a single active component failure. Therefore, no postulated break and a single active failure can totally disable the core spray system. The instrumentation hydraulic lines are also physically separated from each other. The electrical equipment associated with this instrumentation is outside containment Adequate instrumentation is available for actuation of safety systems and plant monitoring following postulated breaks.

The two emergency condenser trains do not have this degree of separation since the supply and return lines for both condensers are located in the same quadrant. The ADS valve controls share a common cable tray which circles the containment. With the separation inside containment and other electrical components located outside containment, the only electrical equipment to be considered is the cable tray with the ADS controls. Thus the ADS and the emergency condenser are the most significant targets in the interaction studies.

D. EVALUATION OF BREAK EFFECTS

It is acceptable under current SEP criteria to use the interaction study to determine the acceptability of plant response to pipe breaks and to assure that pipe breaks would not cause the loss of needed functions of "safety-related" systems, structures and components and to assure that the plant can be safely shutdown in the event of such breaks. The crucial prerequisite is that the interaction study, i.e. the effects of pipe whip and jet impingement on structures and essential safety equipment must have been adequately and conservatively identified and evaluated. Based on a review of Reference 3, we have determined that some areas have not been addressed adequately in the licensee's evaluations, as noted in the discussion below:

1. Dynamic Effects of Pipe Whip

It is not clear that the Oyster Creek pipe break interaction study meets current criteria with respect to the analyses of pipe motions caused by the dynamic effects of postulated failures. It is acceptable to use a simplified, reasonable judgment to determine the most likely, direct path of motion of the ruptured pipe by the piping geometry and configuration when they are not complicated. For example, if the postulated pipe break occurs at an elbow downstream of the energy reservoir which supplies a high energy flow stream to the break area and causes formation of a plastic hinge, the jet reaction force, i.e., the motion of the pipe whip, can easily be determined. However, in the case of a pipe break occurring in the middle of a rather complicated unrestrained piping configuration, it is necessary to investigate all the possible paths for pipe motion. The complicated dynamic effects induced by the impact and rebound of the ruptured pipe should be considered and the extent to which structures and essential safety equipment are affected by the postulated pipe failure should be determined.

The licensee has already considered in Reference 3 the interactions of whipping pipes with other high energy lines and safety systems based on an assumed direction of motion of the broken pipe. No unacceptable interactions were identified in this review.

In order to complete the pipe whip interaction review, one means would be for the licensee to perform the following steps:

- For the 141 break locations, determine which breaks can be categorized as "simple geometries" for which the previously assumed pipe motions can be utilized.
- Of the remaining breaks, in some cases it may be straightforward to determine by inspection that there are no nearby targets regardless of the direction of the pipe whip.
- For the remaining locations determine what systems could be impacted by a whipping pipe.

- In determining if pipe whip in other planes of motion results in interactions not previously identified attention should be focused on the vulnerable targets, such as the emergency condenser.

- If new interactions are defined, demonstrate that consequences are acceptable or provide other justification.

In considering the pipe whip damage, the licensee has used the conservative assumption that the particular safety system becomes inoperative, irrespective of the actual energy which would be involved in the collision or the strength of the target piping system. However, the licensee's evaluation (Table 3 of Section IV, Rev. 3) has identified 16 postulated breaks which could have a secondary effect, i.e., pipe whip could result in contact with other piping, which, if damaged, could initiate further damage. For example, break number 1 in loop C of the reactor recirculation piping will interact with the north main steam line. However, a break in the north main steam line will interact with the north feedwater and control rod drive hydraulic return lines. Therefore, the licensee should expand Table 3 to indicate the full extent of all the possible sequences of events or provide the basis for concluding that there will be no subsequent damages.

2. Containment Wall Penetration

The licensee references Chicago Bridge & Iron Company (CB&I) Test Report, "LOADS ON SPHERICAL SHELLS," (Reference 7) which indicates that when the containment wall is loaded over a large enough area i.e., equivalent to a 14 inch diameter or larger circle, deformation of over 3 inches can occur without failure of the containment wall. Based on this test result, the licensee concludes that for breaks occurring in piping of 14 inch diameter or greater, such as main steam piping, feedwater piping, recirculation loop piping, even if contact occurred with the containment wall, the wall could deform without failure until deformation was limited by the concrete shield wall. Accordingly, the licensee has concluded that break in these lines would not penetrate the containment wall and, therefore, that they would have acceptable consequences.

Based on the information submitted, we have determined that the licensee has not provided a valid rationalization to justify the use of CB&I test results in their case. It is noted that this test was performed under essentially static conditions. It is not clear that the test result is also valid for the dynamic loading which would be experienced as a result of pipe whip. In addition, the particular test applies a concentrated load of 235 tons over an area, equivalent to a 14 inch diameter circle. This assumption may not always be valid because the impact area of a 14 inch diameter or larger pipe may be smaller than the assumed area. Our concern is that in the case of applying a concentrated dynamic load over a small area the steel plate may be perforated before the deformation could be backed up by the concrete shield wall. The probability of perforation could be even higher in those cases where the gap between the liner and the concrete wall is greater than the nominal distance of 2.75 inches. The licensee is requested to provide more justification.

3. Effects of Jet Impingement

Standard Review Plan Section 3.6.2 requires that the jet area expands uniformly at a half angle not exceeding ten degrees and that the total impingement force acting on any cross-sectional area of the jet is time and distance invariant. The licensee's evaluation assumes a jet in the formation of a cone with an angle of divergence of thiry degrees and with the assumption that any safety system inside this cone becomes inoperative, irrespective of the actual jet impingement force. We have determined that the licensee's jet impingement criteria are acceptable because they are conservative with respect to the currently acceptable criteria.

a. Jet Impingement on Electrical Equipment

As discussed above, due to the physical separation and redundancy of electrical equipment within containment, the only target is the cable tray for the ADS valve controls.

In their interaction studies, the licensee determined that breaks in several different high energy systems, ranging in size from a 2" diameter pipe to the 26" diameter recirculation pipe, could produce jets that would contact this cable tray. These breaks correspond to areas from .02 ft^2 up to the largest double-ended breaks considered in the ECCS analyses.

The reference LOCA analyses for Oyster Creek (Reference 11 and 12) assumed operation of five ADS valves, one emergency condenser and low pressure core spray. A summary of the sensitivity of peak clad temperature (PCT) to break size is shown in Table 1. Feedwater, steam and core spray line breaks are included as well as various size breaks of the recirculation system.

The licensee determined by inspection within the containment the path of the cables from the penetration into the drywell to the valve itself. Based on the arrangement of cable routings around the containment, no more than 3 valves could be affected by jets from any break. Therefore, even assuming a single failure, at least two ADS valves, one emergency condenser and low pressure core spray would be available to mitigate the break effects.

For breaks larger than 1.0 ft^2 (13" diameter) no ADS is needed since system pressure is reduced to core spray initiation pressure before the 120 second timer delay on the ADS expires.

For smaller breaks, the ADS aids in the depressurization and thus shortens the time until core spray is initiated.

A smaller break of .35 ft^2 (8" diameter) was reanalyzed by the licensee without ADS or emergency condensers; the calculated PCT was 2025°F. For this event, the reactor depressurizes through the break with no other energy removal or inventory makeup until the low pressure core spray begins. Performance with one emergency condenser and some ADS valves would be better than the case analyzed with no high pressure means of heat removal or depressurization.

Previous sensitivities for break sizes from 0.35 ft^2 to 1.0 ft^2 showed an increase in PCT of 130°F. Since the energy removed through the ADS is less for the larger break sizes, and thus the ADS is less of a factor in determining overall performance, breaks larger than the 0.35 ft^2 break should not result in PCTs in excess of 2200°F, even with no ADS or emergency condensers. Note that a break in emergency condenser piping would fall in this range, so that break, plus a failure of the remaining condenser, would not have unacceptable consequences.

As shown in Table 1, the PCT decreased markedly with decreasing break size from the .35 ft^2 break down to the 0.1 ft^2 (4" diameter). With some ADS and emergency condenser availability, the PCT for these smaller breaks should not exceed 2200°F.

The main steam and feedwater line breaks resulted in substantially lower PCTs than did the recirculation line breaks. The reduction from 5 ADS valves to 2 valves should not result in PCTs worse than the cases discussed above.

For postulated high energy line breaks, the staff concludes that the unaffected ECCS are capable of mitigating the consequences. Since a large range of break sizes have already been considered, assuming different directions of jet motion should not alter these conclusions. The staff concludes that the licensee for Oyster Creek has adequately considered the effects of jet impingement on electrical equipment.

b. Jet Impingement on Mechanical Equipment

The licensee considered the effects of jets on mechanical equipment and piping systems in a manner similar to that done for the effects of pipe whip. The reassessment of interactions with other pipe motions should also include jets to ensure that all potentially adverse consequences have been addressed.

VI. CONCLUSIONS

Based on the information submitted by the licensee, we have reviewed the criteria pertaining to the locations, types and effects of postulated pipe breaks in high energy piping systems inside containment. We have concluded that the criteria used to define the break locations and the types are, in general, in accordance with currently accepted standards. We have also determined that it is acceptable under current SEP criteria to use the interaction study to evaluate the effects of postulated pipe breaks and to determine the acceptability of plant response to pipe breaks. However, we have found that certain areas of the interaction studies, as identified in the "EVALUATION" sections of this report, have not been addressed adequately in the licensee's evaluation and should be resolved.

TABLE I

Break	1 Emergency Condenser 5 ADS Valves (Reference 12)	1 Emergency Condenser 5 ADS Valves (Reference 11)	0 Emergency Condensers 0 ADS Valves (Reference 3)
Small Breaks			
.10 ft^2	1212°F. ADS at 163 sec., rated core spray at 393 sec.	-	-
.35 ft^2	1855°F. ADS at 135 sec., rated core spray at 187 sec.	1960°F. ADS at 137 sec., rated core spray at 189 sec.	2025°F. no ADS
1.0 ft^2	1984°F no ADS, rated core spray at 89 sec.	-	-
Split Breaks			
1.0 ft^2	2012°F.	2076°F.	-
2.5 ft^2	2013°F.	2123°F.	-
Large Breaks			
6.292 ft^2			
C_D = 0.4	2158°F.	2200°F.	
C_D = 0.6	2131°F.	2168°F.	
C_D = 1.0	2060°F.	-	-
Steam Line Break	682°F.	-	-
Core Spray Line Break (.196 ft^2)	1335°F. ADS on, rated core spray at 527 sec.	-	-
Feedwater Line Break (.998 ft^2)	comparable to core spray break; ADS on	-	-

REFERENCES

1. September 7, 1978, "SEP Topic III-5.A, Pipe Break Inside Containment."

2. February 6, 1979, "High Energy Piping Systems Inside Containment, Stress Summary."

3. July 30, 1979, "SEP Topic III-5.A, High Energy Piping Systems Inside Containment, Effects of Postulated Pipe Breaks."

4. Letter from Nuclear Regulatory Commission, SEP Branch Chief to KMC, INC, dated July 20, 1978.

5. Amendment 68 (Supplement 6, Addenda 5) to the Oyster Creek FDSAR, Application for a Full Term License, dated March 6, 1972.

6. M. E. Nitzel, "Summary of the Oyster Creek Unit 1 Piping Calculations Performed for the Systematic Evaluation Program," EGG-ZA-5211, EG&G Idaho, dated July 1980. Performed for SEP Topic III-6, "Seismic Design Consideration.

7. Philip Thullen, "LOADS ON SPHERICAL SHELLS," Oak Brook Engineering Department, Chicago Bridge & Iron Company, dated August 1964.

8. Letter from the Nuclear Regulatory Commission (G. Lear) to Jersey Central Power & Light Company (I. Finfrock), dated February 12, 1975.

9. Revision 1 to Supplement 6 (Addenda 5) to Amendment 68, dated March 6, 1975.

10. Letter from the Nuclear Regulatory Commission (D. Crutchfield) to Jersey Central Power & Light Company (I. Finfrock), dated September 25, 1980.

11. "Oyster Creek LOCA Analyses Using the ENC NJP-BWR ECCS Evaluation Model", XN-NF-77-55, Revision 1, March 1978.

12. "The Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant", XN-75-55, Revision 2, and Supplements 1 and 2, August, September and December 1976.