



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

January 19, 1983

Docket No. 50-237  
LS05-83-01-025

Mr. L. DelGeorge  
Director of Nuclear Licensing  
Commonwealth Edison Company  
Post Office Box 767  
Chicago, Illinois 60690

Dear Mr. DelGeorge:

SUBJECT: SEP INTEGRATED ASSESSMENT STATUS FOR THE  
DRESDEN NUCLEAR POWER STATION, UNIT 2

By letter dated December 6, 1982, you provided your position on the 34 open topics summarized in Chapter 4 of the draft Integrated Plant Safety Assessment Report (IPSAR) for Dresden Unit 2. In addition, the staff has received many other letters regarding specific topics over the past three months. All of the information you provided is being used to finalize the Dresden Unit 2 IPSAR. Publication of the final report is scheduled for January 31, 1983. The purpose of this letter is to provide you a status of all of the open items which will be identified in the final report.

Enclosure 1 contains a list of all the identified differences for which no further action or backfit is required. Included in this list are those topics where (1) the corrective action is complete, (2) the review is covered by another NRC program, and (3) those items which were resolved in the draft IPSAR or as a result of additional information received since the draft IPSAR.

Enclosure 2 lists those issues for which you have committed to implement hardware modifications to the facility. For those items identified in Enclosure 2, the staff will require a schedule for completion of the modifications within 30 days of receipt of this letter.

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Those issues for which you have committed to make procedural or technical specification (TS) changes are listed in Enclosure 3. The procedural changes will be reviewed by the NRC Region III Office. It is the staff's understanding that all of the procedural changes and the proposed TS changes will be completed by the end of the current refueling outage. If the actual schedule differs from our assumption, please inform the staff within 30 days of receipt of this letter.

Enclosure 4 lists those issues which require additional information or analysis. In some instances, you have provided further information which is undergoing staff review. Those items are noted in the enclosure. However, the majority of the items are still outstanding. In addition, you have responded to a few items and the staff review found the information to be insufficient. These items are described in Enclosure 5. Of special note is that regarding SEP Topic III-2, "Ventilation Stack." As described in the enclosure, this issue is being reopened due to your evaluation being performed using a methodology not accepted by the staff. Please review the two enclosures and provide your schedules for completion of the required information within 30 days of receipt of this letter.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you have any questions regarding the enclosures, please contact the Dresden Unit 2 Integrated Assessment Project Manager, Greg Cwalina, at 301-492-8053.

Sincerely,

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

Enclosures:  
As stated

cc w/enclosures:  
See next page

High pressure coolant injection piping, fittings and valves - These components are made of A106, Grade B carbon steel of various thicknesses. The drain and condensate lines are exempt from testing due to thickness considerations. The piping requires testing, but may be exempt from testing if the licensee confirms that the lowest service temperature, as defined in the ASME Code, is greater than 150°F. Otherwise, the licensee must demonstrate the adequacy of the fracture toughness for this component or demonstrate that the consequences of failure of the component are acceptable.

- (d) Condensate/feedwater system piping - This piping is A106, Grade B, with thickness ranging from 0.718" to 1.093". This piping requires testing, but may be exempt from testing if the licensee confirms that the lowest service temperature is greater than 150°F. Otherwise, the licensee must demonstrate the adequacy of the fracture toughness for this component or demonstrate that the consequences of fracture of the component are acceptable.
- (e) Main steam system - This system is made of A106, Grade B carbon steel and is 1.031" thick. This piping requires testing, but may be exempt from testing if the licensee confirms that the lowest service temperature is greater than 150°F. Otherwise, the licensee must demonstrate the adequacy of the fracture toughness for this component or demonstrate that the consequences of fracture of the component are acceptable.

Since you have not provided the necessary information, the staff is unable to conclude that adequate fracture toughness exists. Further, you have not supplied any information regarding radiography requirements. Therefore, the staff position identified in the draft IPSAR remains unchanged. The staff will require that the necessary information be supplied in a revision to the updated FSAR within two years.

5.22 Question

Pipe whip criteria that have been implemented in the design of the plant should be stated. A description of how these criteria have been used in the design of various engineered safety features should be included, with particular emphasis on ECCS piping and instrumentation systems located within the drywell. The description should also indicate how the effects of jet impingement forces on various safety feature components have been accounted for in the design.

Answer

Pipe restraints to prevent pipe whip have been applied where deemed necessary to insure that:

- a. containment integrity will be maintained,
- b. at least one core spray system, including instrumentation, will remain operable, and
- c. at least one set of reactor pressure vessel level instrumentation will remain operable.

It is felt that this criteria has been met by:

- a. the application of pipe restraints to the recirculation loop,
- b. physical separation of redundant ECCS piping and instrumentation, and
- c. physical separation of level instrumentation.

Similar criteria has been satisfied under jet impingement forces through containment and penetration design, and the physical separation of ECCS components.

QUESTION

I. A For certain of these items, the analyses, research and development, and design changes to provide resolution have been adequately described. However, further technical information is needed regarding the specific actions taken to provide adequate resolution of those items listed below:

1. Jet pump operation, monitoring, and system stability
2. Pipe whipping and missile generation
3. Pressure vessel design, with attention to bell-mouthing and vibration
4. Independent review of vessel stress report
5. Periodic vessel inspection
6. In-core flux monitoring instrumentation
7. Conservatism in design and fabrication of the primary system
8. Core analytical models
9. Load control with variable speed pumps
10. Dresden lock and dam failure

ANSWER

Please note that items 1, 7, 8, 9 and 10 are answered in Questions I. B, I. C, I. D, I. E, and I. F, respectively.

2. Section 5.2.3.7 of the FSAR discusses the analyses which were made on potential missile generation inside the containment. The analyses show that there are no sources of missiles within the drywell with the energy potential required to penetrate the containment shell.

The recirculation lines have been provided with restraints to limit the motion of these lines. The restraints are discussed in Section 4.3.2 of the FSAR. The restraints will limit any motion of the recirculation lines during a postulated but highly improbable recirculation-pipe rupture.

The examination of other piping systems within the drywell has led to the conclusion that the main steam lines and the feedwater lines contain sufficient energy, should one of these lines suffer an instantaneous complete severance of the pipe in certain specific locations, that the broken line could possibly penetrate the containment shell. Therefore, studies have been made and tests have been conducted to determine the failure mode of this piping, i. e., to determine if the piping can sever completely and in a short enough time period to develop the energy that is required to penetrate the containment shell.

Tests have been conducted as part of the AEC sponsored Reactor Primary Coolant Rupture Study which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate rapidly. The applicability of this study to the evaluation of the problem of pipe rupture in the drywell has been discussed in detail in Oyster Creek, Docket 50-219, Amendment 34. The results of the tests indicate that for a crack of a size which gives a leakage of 5 gpm the probability of rapid propagation is  $10^{-6}$ . Thus, a pipe which is cracked and for which the leak rate is approximately 5 gpm, there is a probability of one in one

million times that the crack will propagate rapidly enough to result in complete severance of the pipe. If the crack were of the size that resulted in a leakage of 15 gpm, there would be a probability of rapid propagation of  $10^{-4}$ . This crack size and leakage rate is well within the leak detection capability provided for the drywell. The leak detection capability in the drywell is discussed in Section 4.3.4 of the FSAR.

The conditions for critical (or unstable) crack growth are based on the assumption that the cracks grow to critical size by mechanically or thermally induced cyclic loading or stress corrosion cracking or some other mechanism characterized by gradual crack growth. From the tests conducted and the rate of crack growth it can be concluded that the main steam lines and the feedwater lines will not suffer complete severance during the lifetime of the plant. Therefore, restraints cannot be justified for these piping systems.

If an undetected fault could lead to rapid propagation or complete severance of any pipe in the drywell, it would occur in a small diameter pipe. There are no failures which can occur in small diameter pipes which will lead to a penetration of the containment.

Studies that have been done for the Oyster Creek Plant to investigate the manner in which pipe restraints would have to be designed to provide complete restraint capability for the steam and feedwater lines similar to the restraint capability that has been provided in the Dresden 2, 3 plant design for the recirculation system piping have shown that it is impractical to design such restraints for the steam and feedwater piping because of mechanical and structural limitations from the point of view of anchoring these specific lines. In addition, the conceivable restraining devices that would be installed would have to be installed in such a manner that the restraints would prevent convenient and careful inspection of the sensitive sections of these pipe lines without removing the mechanical restraint devices at each inspection period.

The Dresden Unit 2 and 3 main steam and feedwater piping and the drywell structural configurations are essentially similar to those of Oyster Creek and the problems of installing restraints are comparable to those encountered in the Oyster Creek investigation.

Therefore, it is our belief and recommendation that preventive maintenance and regular inspection of sensitive pipe runs is a more safe method to be followed in assuring that large pipes in the reactor drywell will not fail, rather than assuming that very unlikely failures can occur and providing massive restraining equipment that in themselves compromise the opportunity to perform maintenance and inspection activities.

In order to provide the maximum assurance that the emergency core cooling system piping and instrumentation will perform the required functions in the unlikely event of a pipe rupture within the drywell, these systems have been physically separated

and made redundant such that a postulated failure of any piping in the drywell will not disable or prevent proper operation of the emergency core cooling system. In addition, where possible, ECCS piping has been routed behind structural members for additional protection of the ECCS system.

3. Bell-mouthing of the reactor vessel is applicable only to vessels with breech closure, or a closure made by screwing the reactor vessel head into the reactor vessel. This type of closure is not used on the Dresden 2,3 reactor vessels therefore bell-mouthing is not applicable.

The Dresden 2,3 reactor vessels were designed and built in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The reactor vessel internals were designed with attention being given to vibrations and the reactor vessel and internals were analyzed to determine their capability to withstand flow induced vibrations. Vibration measurements will be made, during the startup test period on the Dresden 2 reactor vessel and internals to demonstrate the mechanical integrity of the system to vibration motions. These measurements are also designed to check the validity and accuracy of the analytical procedures used to calculate the vibration characteristics of the system. The following points will be monitored for vibration:

- a. Control rod guide tubes.
  - b. In-core guide tubes.
  - c. Fuel channels.
  - d. Core plate.
  - e. Shroud.
  - f. Separators.
  - g. Recirculation loops.
  - h. Jet pumps.
4. The vessel stress report is being prepared by the Babcock and Wilcox Company. An independent review of this vessel stress report is conducted by the General Electric Company on each section of the report as it is received from B and W. Upon completion of the analysis and the G. E. approval, a certified report will be issued.
  5. Details of the periodic inspection program for the reactor vessel and primary system piping are contained in the Technical Specifications, Section 4.5.
  6. A detailed report of the in-core flux monitoring instrumentation was submitted as topical report APED-5706, December 1968.