Leakage of primary coolant at Mihama Unit 2 due to failure of SG tube

February 9th, 1991, Mihama in Fukui Prefecture

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(Summary)

On February 9th, 1991, a heat transfer tube (SG tube) in a steam generator of the No.2 pressurized water reactor at the Mihama nuclear power station of the Kansai Electric Power Company broke off during a rated output operation. As a result, about 55 tons of primary cooling water leaked out from the SG tube into the secondary cooling loop, and the reactor was scrammed by operation of the ECCS (Emergency Core Cooling System). The failure of the SG tube was caused by fretting fatigue resulting from contact of the SG tube with the supporting plate for the SG tubes, because the AVB, which functions to prevent flow-induced vibration, was not inserted deep enough onto the SG tubes in the steam generator. The scale of the accident was ranked "level 3" on the international nuclear events scale (INES).

1. Component

Steam generator of pressurized water reactor

Figure 1 shows a schematic illustration of the pressurized water reactor. Figure 2 shows the steam generator and the location of the failure of the SG tube.

The steam generator is a pressure vessel with a heat exchanger. The steam generator has an outer diameter of around 4 m and a height of around 20m. 3260 SG tubes were installed in the heat exchanger. The tubes, each with an outer diameter of 22.2 mm and a thickness of 1.27 mm, were made of Inconel 660 material. Most of the length of the SG tubes was straight, but the upper part of the tubes in the heat exchanger was bent in an opposite U-shape with certain curvatures. The straight part of the SG tubes was held at six positions by supporting plates. At the sixth supporting position near the upper part of the SG tubes, a specifically shaped (V-shaped) anti-vibration bar, AVB, was installed on the bent part of SG tubes to prevent flow-induced vibration.

2. Event

At around 13: 50, on February 9th, 1991, leakage of about 55 tons of primary coolant occurred due to the failure of a SG tube in a steam generator of the No.2 pressurized water reactor at the Mihama nuclear power station of the Kansai Electric Power Company. The reactor shut down commenced immediately triggered by a signal warning that the primary coolant in the reactor was decreasing. The ESSC was operated immediately, and about 51 tons of coolant water was flooded into the reactor. The amount of steam released from the main steam relief valve to atmosphere was about 1. 3 tons. The amounts of radioactive rare gas and iodine discharged to the atmosphere were about 2.3E10 and 3.4E8 becquerels, respectively.

After the accident, investigation of the steam generator was carried out using a fiber scope and some other inspection instruments. As a result, the failure of a SG tube was found near the sixth supporting plate for the SG tubes.

3. Course

At 13:40, an alarm of a condenser air off take system went off during a rated output operation, warning that the coolant water level in the steam generator was decreasing. At 13:50, an automatic emergency shutdown of the reactor was triggered by the signal of decreasing pressure in the pressurizer. After seven seconds, the ECCS was automatically operated, and coolant water was flooded into the reactor by a high pressure injection pump. However, one main steam isolation valve and one pressurizer relief valve could not be operated by remote control. Therefore, the valve operation was carried out manually.

4. Cause

(1) Fracture surface

The failed tube was removed from the heat exchanger, and the fracture surface was examined by a scanning electron microscope. Striations, which are a characteristic of fatigue failure, were observed on large portions of the fracture surface, and dimples showing tensile fracture were also observed. However, few traces of stress corrosion cracking and corrosion were found on the fracture surface of the tube. The failure of the tube was, therefore, hypothesized to be due to cyclic loading. Figure 3 shows the morphology of the fracture surface of the SG tube. Figure 4 shows a typical example of the striations formed on the fracture surface of the SG tube. Examination of the other SG tubes near the failed tube showed traces of wearing formed by fretting due to contact between the tubes and the anti-vibration bars on the outer surfaces of the tubes. Stress amplitude of the failed tube estimated based on the striation spacing was found to be in the range of around 51 to 60 MPa.

(2) Fatigue fracture

Occurrence of cyclic loading in the SG tube that had failed was related to the insertion depth of the anti-vibration bar, AVB. The SG tubes were subjected to vibrations due to the flow of secondary coolant outside of the SG tubes. In order to prevent the flow-induced vibration, V-shaped AVBs were installed onto the opposite U-bent SG tubes near the upper part of the steam generator. However, the insertion depth of the AVB for the SG tubes was not enough, because the engineers who installed the AVB did not fully understand the importance of the AVB. In fact, no damage was founding in the SG tubes into which the AVB were inserted to sufficient depth as shown by the design guidelines. Figure 5 shows the insertion position of the AVB on the SG tubes by solid lines. Accordingly, the SG tubes were subjected to flow-induced vibration and strongly contacted with the sixth supporting plate, so that the SG tubes incurred damage by fretting fatigue. Inspection of the AVB had not been carried out since installation. Figure 6 shows a schematic illustration of the fatigue failure of the SG tube due to flow-induced vibration.

5. Immediate Action

After the accident, a detailed examination of the AVB for the SG tubes in the steam generators of all reactors in the Kansai Electric Company was carried out. As a result, lack of sufficient insertion depth of the AVB was found in some heat exchangers. These AVBs were replaced with new ones and installed at the designated depth in the steam generators. Moreover, the steam generator of the No.2 pressurized water reactor was also replaced with a new one, because many of the SG tubes were removed from the generator for failure analysis.

6. Countermeasure

(1) Inspection of the installation of the AVB for the SG tubes and of the mounting position of the AVB before operation.
(2) Execution of regular inspections of the AVB.
(3) Inspection for distortion and damage of the supporting plate for the SG tubes.
(4) Development of a new detecting system that can quickly and accurately detected signs of damages of the SG tubes in a steam generator.
(5) Development of a new type of AVB with high performance and easy installation.
(6) Employers should make sure that the engineers who are engaged in fabrication or maintenance of the devices and equipment in nuclear reactors understand the function of those devices and equipment.

7. Knowledge

A discriminate equation describing the generation of flow-induced vibrations for SG tubes was suggested based on a maintenance standard of the Japan Society of Mechanical Engineers, as follows.

$$SR = Ue/Uc < 1$$

Where SR is the discriminate value, Ue is the effective velocity, and Uc is the critical velocity.

8. On the Side

In order to provide an opportunity to learn from the accident that resulted in the leakage of primary coolant from the SG tube due to fretting fatigue, the damaged steam generator has been preserved in an exhibition at the Mihama station of the Kansai Electric Power Company. An exhibition is a good way to help everyone to good lessons from an accident.

9. Social Impact

This accident was the first disaster in Japan that resulted in actuation of the ECCS due to leakage of primary coolant in the steam generator. Therefore, the accident caused social concern with nuclear reactors.

The international nuclear events scale (INES) is defined by the IAEA to assure coherent reporting of nuclear accident by different official authorities. The INES is characterized from level one to level seven. The level number increases with the scale of the accident. For example, level one is a minor event, and level seven is major accident. The scale of the accident in 1979 resulting in the loss of coolant that occurred in Three Mile Island was ranked level five by the IAEA. The accident reported here was ranked level three.

10. Information Source

(1) Nuclear Power Engineering Test Center, Nuclear Power Safety Information Research Center, On the damage of SG tube in steam generator of No.2 pressurized water reactor at Mihama Station of Kansai Electric Power Company, (1992)

(2) htt p://www.atom.meti.go.jp/atom-db/jp/index.html

(3) Maintenance standard of the Japan Society of Mechanical Engineers, JSME S016-2002

11. Primary Scenario

01. Ignorance
02. Insufficient Knowledge
03. Poor Understanding
04. Production
05. Hardware Production
06. Production of Machinery and Equipment
07. Heat Exchanger
08. SG tube
09. Supporting Plate
10. AVB
11. Installation
12. Regular Operation
13. Nonobservance of Procedure
14. Error of Mounting Position
15. Lack of Insert Depth
16. Failure
17. Fracture/Damage
18. Flow-induced Vibration
19. Fretting Fatigue
20. Failure of SG tube
21. Usage
22. Maintenance/Repair
23. Inspection
23. Lack of Inspection
24. Failure
25. Large-Scale Damage
26. Leakage of Coolant

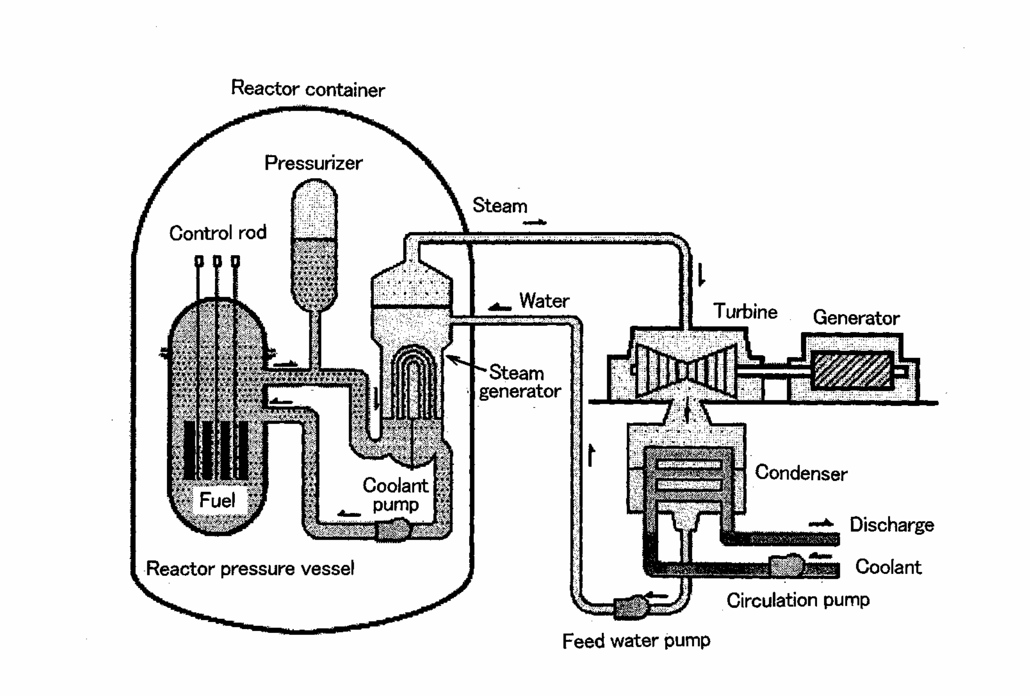


Fig. 1 Schematic illustration of pressurized water reactor.

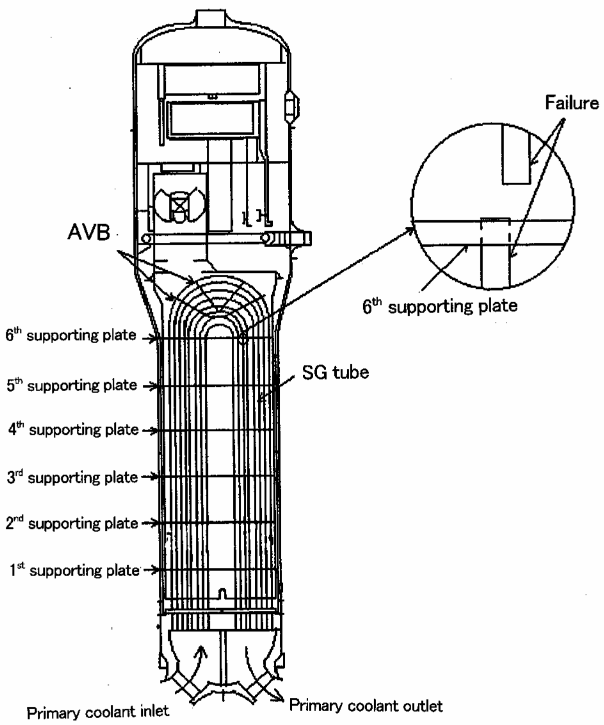


Fig. 2 Steam generator and location of failure of SG tube.

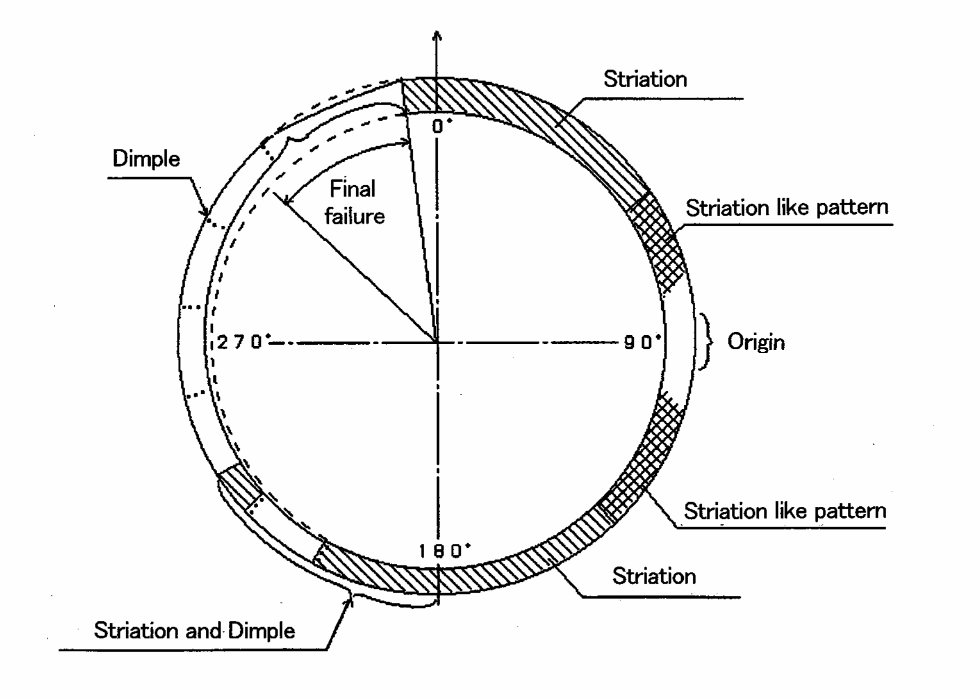


Fig. 3 Morphology of fracture surface of the SG tube.

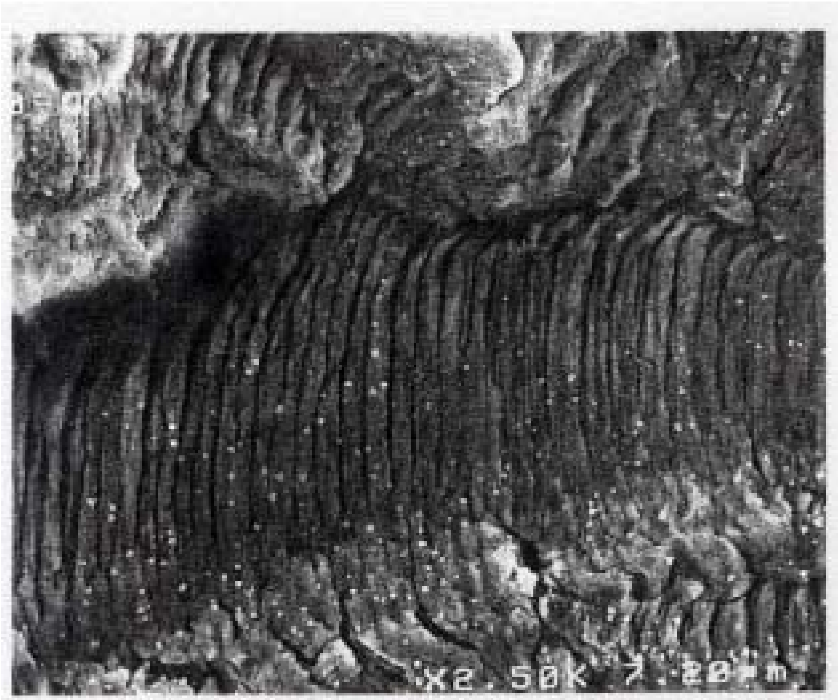
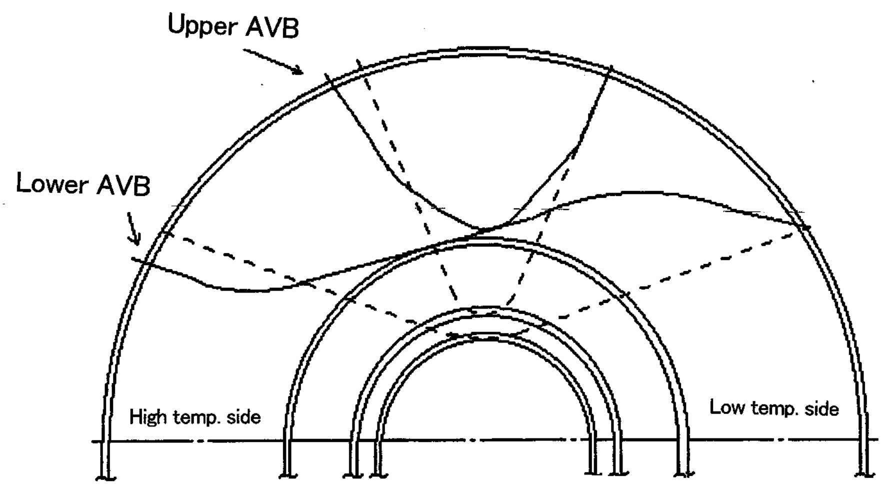


Fig. 4 Striation formed on the fracture surface of the SG tube.



Solid lines show the location of AVB when accident occurred.

Dotted lines show normal position of the AVB.

Fig. 5 Location of AVB on the SG tubes.

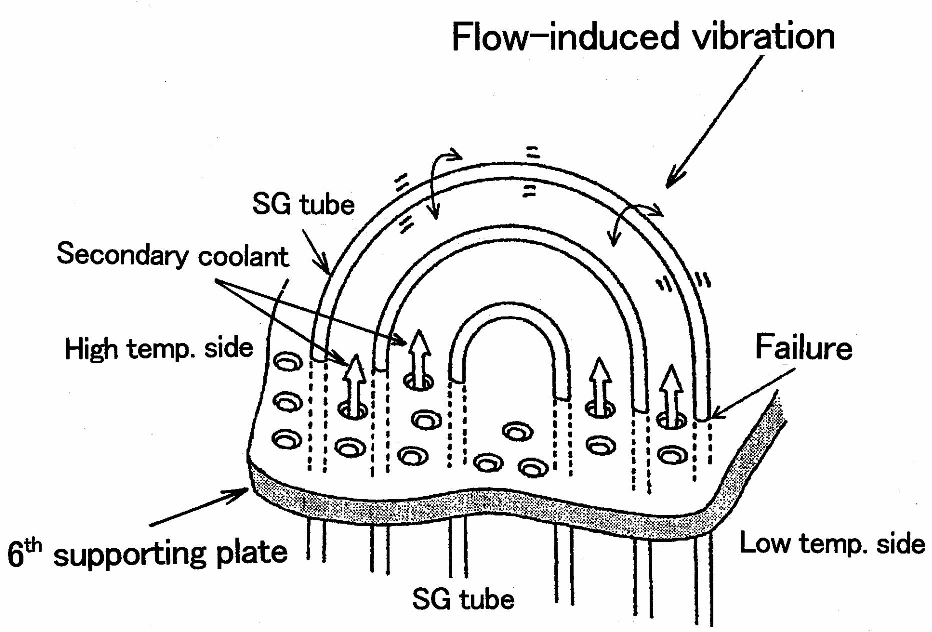


Fig. 6 Schematic illustration of fatigue failure of the SG tube due to flow-induced vibration.