高速炉トラブル事例の着目項目(文献から抽出したい情報)

① 当該施設名(高速炉プラント、試験施設の名称)

② トラブルの発生日時

③ トラブル発生箇所(系統、機器名称 等)

④ トラブルの内容及び状況
- 冷却材の漏えい等のトラブルの内容及び事象推移
- 破損の規模(亀裂寸法等)
- 漏えい事象の場合、漏えい量/漏えい速度

⑤ トラブル原因(溶接不良、熱応力による亀裂進展 等)

⑥ トラブル発見方法
(漏えい監視設備による検知、運転員による発見 等)

⑦ トラブル発生時のプラント/施設の運転状態
(プラント起動操作中、定格運転中、停止中 等)

⑧ トラブル後の対応
- 事象発見直後の対応
- 補修等の再稼働に向けた対応(補修方法、水平展開、再稼働時期)

文献調査のイメージ（DFRの冷却材漏えい）

Location and repair of a leak in the Dounreay Fast Reactor Primary Circuit*

Investigation and location of the leak

R. R. Matthews, MA, FIMechE, FIEE, AMIChemE

Repair of the primary circuit pipework

K. J. Henry, BSc, MIMechE

- Dounreay Fast Reactor
- Primary Circuit

THE CHAIRMAN: This evening we have a most interesting topic for discussion, the location and repair of a leak in the Dounreay Fast Reactor Primary Circuit. We are very fortunate to have with us Mr Matthews, a Director of the Reactor Group, UKAEA, who at the time was Director of the Dounreay Experimental Reactor Establishment.

We also welcome Mr Henry of the Fast Reactor Design Office. He was also with the Dounreay Establishment at the

cuit could be repaired, and no special provision was made for doing so. During the design phase there was emphasis on safety and prevention of loss of coolant, so all primary circuits were enclosed in a closely fitting leak jacket, and it was this leak jacket which made location of the leak extremely difficult.

When we knew on 9 May, 1967, that a leak had been detected I think we all felt some apprehension about the

there was very little knowledge and experience of sodium equipment and, consequently, the layout was very different from the more sophisticated designs which are now being used for larger fast reactors. It was then thought to be extremely unlikely that a leak in the vessel or primary cir-

* Report of meeting held at the Institution of Civil Engineers, Great George Street, Westminster, London SW1, on Thursday, 20 February, 1969. Professor G. R. Hall was in the chair.

separated from the one with which we are concerned.

At the bottom of Fig. 2 can be seen a drainpipe which leads to a tank outside the vault, and to this is fitted a vertical tube which contains a spark plug type leak detector.

On 9 May, 1967, this detector gave an alarm, and immediately the levels in the expansion tanks and the reactor vessel were checked by the instrumentation. There was no change in level, so it was evident that a gross leakage was

1967年5月9日に漏えい検出器の警報により検知

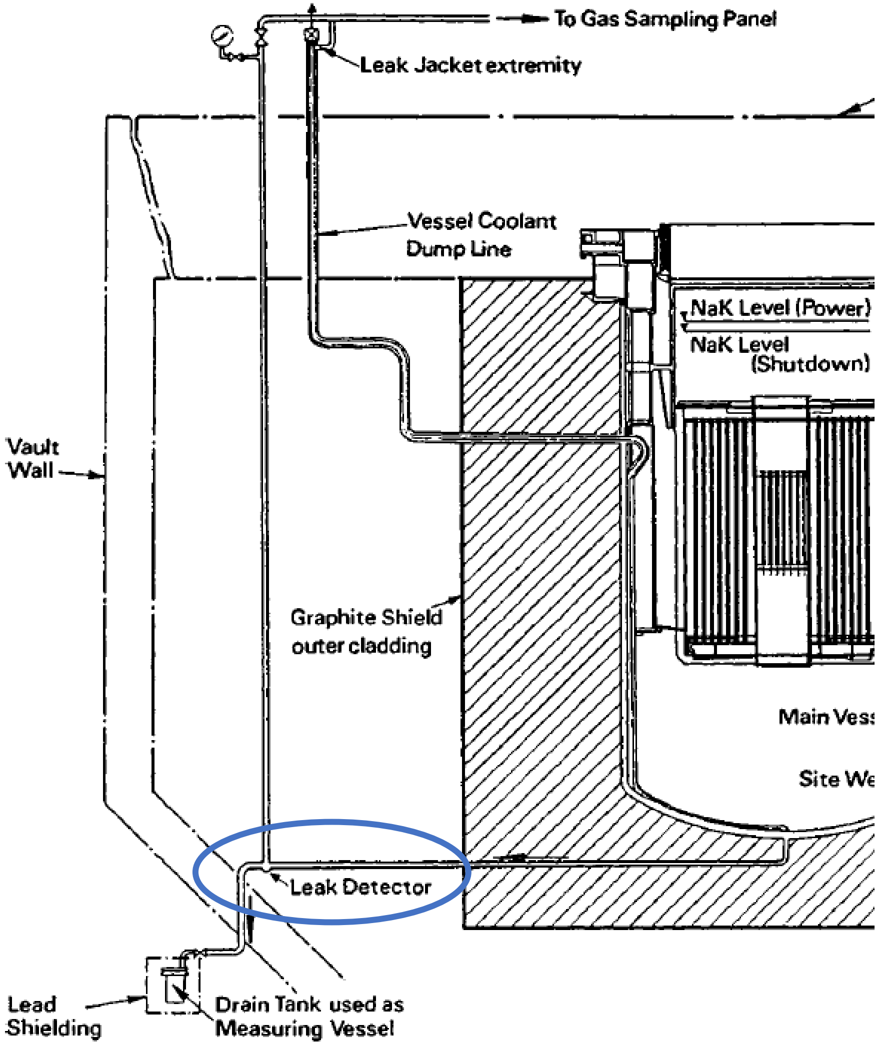

Fig 2 Simplified diagram of the reactor vessel showing only one of the

文献調査のイメージ（DFRの冷却材漏えい）

漏えい確認後通常の手順で停止

not taking place. Also there was no change in the pressure of the argon gas atmosphere in the leak jacket, again indicating no large leakage.

At first it was thought that there might be a spurious instrument fault, but checking showed that this was not the case. Soon afterwards the drain tank was monitored and it was found to contain radioactive coolant, confirming that a leak existed. The reactor was shut down by the normal procedures, a sample was taken from the tank, and on analysis the coolant was found to contain fission products. It was evident from the analysis that the fission products were quite old, although the coolant had been in the presence of a neutron flux up to the time of shutdown. This indicated that the coolant had probably entered the leak jacket, remained in the vicinity of the neutron flux for some time and had then worked its way down the pipe into the drain tank. This indicated that the leakage, which was only a few litres, had taken place during two or three months, and meant that the leak was extremely small.

The reactor vessel was kept under pressure, but there was no evidence of further leakage taking place. As the leak

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運転を継続し、漏えい量の推移を監視

rate was very low, it was decided that the reactor should continue operation and further measurement of the leak rate made, and it would continue operating provided the leak rate did not increase significantly. It was not considered possible to take any remedial action at that stage, because the leak rate was evidently so small.

After carrying out some modifications to the leak jacket system by fitting additional instrumentation, additional leak detectors, and providing means of dumping the coolant from the drain tank during operation, the reactor was started up again and it continued for several weeks through June and July.

The leak reappeared immediately after start up and more coolant came down into the tank. It seemed to steady off at a rate of about 100 l/day, and remained steady for several weeks. The reactor continued in operation, but periodically the drain tank had to be emptied and the radioactive coolant pumped away to a dump tank.

The leak rate increased towards the end of July, and it went up to about 150 l/day, where it again remained steady. Reactor operation continued, but on 29 July the leak rate

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その後漏えい量が100、150リットル/dayに増加

文献調査のイメージ（DFRの冷却材漏えい）

Detailed examination of the failed weld showed that there were some defects in the vicinity of the crack. First, the alignment of the pipe to the spigot of the block was badly out, there being a 0·049 in. misalignment between the two where the wall thickness was 0·080 in., giving a considerable step change in contour at this point. Also the weld tended to run off the line of the joint, and there was a lack of penetration giving rise to a stress raiser where the weld had a double start. The thermal stress was calculated to be within normal limits, but it is probable that, without the stress, failure would not have been caused by these defects alone. The 1 in. connexion shown in Fig. 5 was provided to return coolant from one of the hot traps, and under full power operating conditions there had been a 100 degC temperature difference between the coolant in the 1 in. pipe and that in the 4 in. pipe. This caused a thermal stress which disappeared at reactor shutdown, and so explained why the leak then stopped.

I think you will agree it was a very minor fault which caused all the trouble, and it does point to the necessity for good design and good quality control in fabricating this type of plant. Decisions on the repair work were then

溶接溶け込み不良、溶接の開始点が二重になっている箇所で応力集中が発

文献調査のイメージ（DFRの冷却材漏えい）

Location and repair of the DFR leak

By
R. R. Matthews and K. J. Henry, [U.K.AEA, Dounreay

The location and repair of a small leak in the primary coolant circuit of the Dounreay Fast Reactor constitutes an impressive feat in remote operations

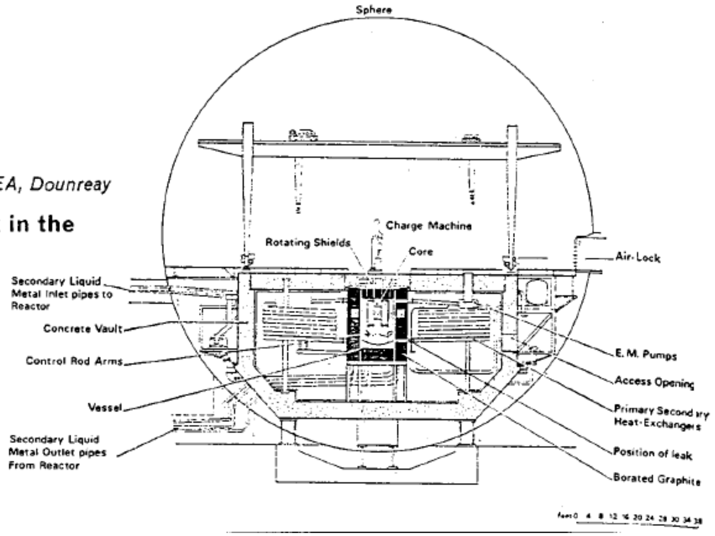

Fig. 1 Section through the containment sphere and the primary circuit vault of the DFR

hacksaws. With this section removed it was p examine the suspect weld visually from a distanc using an introscope and by ultrasonic techniques; evidence of irregularities in the weld bead but loc leak tests proved negative. Meanwhile the 10 ft. se which contained some residual radioactive NaK, was decontaminated and pressure tested. This immediately revealed a crack about 1¼ in. long on the circumferential weld joining the 4 in. pipe to the thermocouple block, in line with and about an inch from a 1 in. diameter "tee" connection to the 4 in. pipe. The leak rate was measured and found to be of the same order as that measured during the circuit leak tests, but was still a large factor down on that needed to account for the leakage of about 200 litres of NaK per day during the last days of reactor operation.

The possibility of the leak rate being affected by temperature gradients and the resulting stresses was investigated. The 1 in. connection was provided to return coolant from one of the hot traps which were fitted at a late stage in the construction programme in order to reduce the oxide content of the NaK coolant to very low levels. In practice they have never been used, as a sufficiently low oxide content has always been obtained by using cold traps alone.* Under full power operating conditions there had been a 100°C temperature difference between the coolant in the 1 in. pipe and that in the main 4 in. pipe. A thermal stress of the same order was applied by cooling the tee junction to a temperature 100°C below that in the main pipe by immersing the former in liquid nitrogen. This resulted in an immediate and reversible increase in the leak rate by about a factor of 40. An adequate explanation had therefore been found for the differing behaviour of the leak between

1¼インチの周方向亀裂

文献調査のイメージ（DFRの冷却材漏えい）

"LEAK BEFORE BREAK" OPERATING EXPERIENCE

FROM EUROPEAN FAST REACTORS

L. MARTIN (1)
J. DUBOUIS (2)
J.D.C. HENDERSON (3)
W. KATHOL (4)

B - DFR

Date	Leak location	Fluid	Leakage	Detection method	Cause
July 29, 66	Secondary heat exchanger superheater	NaK II		Leak detector alarm	Crack in a weld
May 9, 67	Primary heat exchanger pipe work	NaK I		Leak detector alarm	Fatigue failure
June 20, 68	Thermal syphon Primary circuit	NaK II		Operator	Corrosion
June 25, 68	Thermal syphon Primary circuit	NaK II		Operator	Corrosion
August 21, 70	Secondary heat exchanger	NaK II		Operator	Welding defect
Apr. 4, 78 (2)	Thermal syphon Primary circuit	NaK II		Operator	?
Feb. 25, 79 (2)	Liquid metal charge station	NaK II		Operator	?

C - PFR

Date	Leak location	Fluid	Leakage	Detection method	Cause
Mar. 74 (1)	Secondary circuit n° 1 pipe from pump to heat exchanger	Na II		Leak detector	Defect in repaired weld
Jan. 16, 76	Secondary circuit n° 3 reheater	Na II		Noise - tube vibration	Failure of tube/ tube plate welds
June 29, 77	Secondary circuit n° 2 superheater	Na II		Slight leakage of sodium	Fatigue

文献調査のイメージ（EBR-Iの冷却材漏えい）

EBR-I AND EBR-II OPERATING EXPERIENCE

M. Novick
F. D. McGinnis
G. K. Whitham

Argonne National Laboratory
NRTS, Idaho

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anomalies were
tion of that load-
II was much
0.00304), the
h) that it was
l day-to-day mis-
ct that such an
s loading, which
to confirm that
associated with

proach to full power (nominally 1200 kW) from cold clean critical was carried out in the following sequence: 510, 878, and 1193 kW. Following each incremental change in power, a complete set of transfer function measurements was carried out. In general, oscillation frequencies ranged from 0.02 to 10.0 cycles/sec.

To study the effects of coolant inlet temperature on the kinetics of the system, transfer function measurements were carried out at inlet temperatures ranging from a low of 50°C to a high of 230°C. Because of a limit placed on the maximum fuel temperature, 450°C, it was not possible to operate at full power under high inlet temperature conditions. The highest power level attained for an inlet temperature of 230°C amounted to only 930 kW. Nevertheless, sufficient informa-

Component Performance

The primary NaK system, contained in shielded cells, is an all-welded system. No leaks have occurred in the primary system during the life of the facility. The secondary NaK system uses standard 300-psi flanged joints with solid stainless steel O-rings. One leak occurred in this system after several years of operation. Only external cleaning and retightening of the flange were required to correct it.

No failures of any of the steam system, including the NaK-to-water heat exchanger, have occurred, and only normal maintenance of the instrumentation and control system was required. All components may be isolated by double-bellows sealed valves. Such valves are also used to control coolant flow, both from the pumps and to the reactor. Two 4-in. valves in the reactor inlet line and one 2-in. valve on the shutdown cooling loop of the primary system have each been operated approximately 10,000 cycles. No failures have occurred. This is credited to the extensive bellows testing program which was conducted at Argonne prior to specifying the bellows to be used.

Difficulty was experienced with the electromagnetic pump during initial operation. This was caused by gas entrainment, and the installation of anti-vortexing plates in the pump receiver tank and of a means of venting the suction line to the pump eliminated this problem. On one occasion,

after the system was shut down and drained of NaK coolant for an extended period, high tube temperatures were observed upon starting up the system. The pump was operated at reduced load for several hours, and the tube temperature gradually returned to normal. This has been the only instance of abnormal operation since installation in 1951.

The reactor tank has accumulated an integrated irradiation dose of about 10^{20} nvt (>1 Mev) which will have increased the tensile strength about 25% and the hardness about 20% with an accompanying decrease in ductility. This is considered to be in the lower region of known radiation damage, and it is estimated that the tank could be used for continued operation, if necessary.

sodium. Two mechanical pumps, rated at 5,000 gpm each, pump 700°F sodium to the reactor, from which it leaves at 883°F. The hot sodium transfers its heat to the secondary sodium system in a shell-and-tube heat exchanger which is also located in the primary sodium tank. A 6,000-gpm electromagnetic pump in the secondary sodium system pumps non-radioacti
ary heat exchanger into
produce 196,000 lb/hr o
850°F. This steam is u
of electricity which su
network.

テキストをコピー

赤色：2次NaK系での漏洩
二次NaK冷却システムでは、標準的な300psiフランジ接合部と固体ステンレス鋼製Oリングを使用している。このシステムでは運用開始後数年を経て初めて漏水が発生したが、必要な対応はフランジの外部清掃と再締め付けのみで済んだ。

黄色：電磁ポンプのトラブル
システムの初期運転時には、電磁ポンプにおいて問題が発生した。これはガス巻き込みが原因であり、ポンプ受油タンク内に抗渦流防止板を設置するとともに、ポンプ吸込管の排気機構を整備することでこの問題を解決した。

緑色：配管温度の異常上昇
一度、システムを長期間停止してNaK冷却材を排出した後に起動した際、配管温度が異常に上昇する現象が確認された。ポンプを数時間にわたり低負荷運転した結果、チューブ温度は徐々に正常値に戻った。これが1951年の設置以来、唯一の異常な運転事例である。