

FREQUENCY AND DISTRIBUTION OF LEAKAGES IN STEAM GENERATORS OF GAS-COOLED REACTORS

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Abstract

In gas cooled reactors with graphitic primary circuit structures - as HTR, AGR or Magnox - the water ingress is an event of great safety concern. Water or steam entering the primary circuit react with the hot graphite and carbonoxid and hydrogen are produced. As the most important initiating event a leak in a steam generator must be taken into account. From the safety point of view as well as for availability reasons it is necessary to construct reliable boilers. Thus the occurrence of a boiler leak should be a rare event.

In the context of a probabilistic safety study for an HTR-Project much effort was invested to get informations about the frequency and the size distribution of tube failures in steam generators of gas cooled reactors. The main data base was the boiler tube failure statistics of United Kingdom gas cooled reactors. The data were selected and applied to a modern HTR steam generator design. A review of the data showed that the failure frequency is not connected with the load level (pressures, temperatures) or with the geometric size of the heating surface of the boiler. Design, construction, fabrication, examination and operation conditions have the greatest influence on the failure frequency but they are practically not to be quantified.

The typical leak develops from smallest size. By erosion effects of the entering water or steam it is enlarged to perhaps some mm^2 , then usually it is detected by moisture monitors. Sudden tube breaks were not reported in the investigated period.

As a rule boiler leaks in gas cooled reactors are much more rare than leaks in steam generators of light water reactors and fossil fired boilers.

1. INTRODUCTION

Water ingress is of particular significance with regard to safety in gas-cooled reactors with graphitic primary circuit structures such as HTR, AGR or Magnox. Water or steam penetrating into the primary circuit reacts with the hot graphite forming carbon oxides and hydrogen. The most important initiating event for water ingress is a leakage in the steam generator.

In connection with the probabilistic safety study for an HTR project /1/ investigations were made in order to obtain information about leakage frequency and leak size distribution in steam generators of gas-cooled reactors. The main data base was British gas-cooled reactors. For these reactors (AGR, Magnox) extensive, processed data material is available on steam generator failures up the year 1979 chiefly relating to Magnox reactors /2/, /3/. In addition, failure data were available for three helium-cooled reactors. The data were reviewed and checked for transferability to modern HTR steam generators. The selected data were used to derive frequencies for various leakage spectra.

2. OPERATING DATA OF GAS-COOLED REACTORS

The gas-cooled reactors longest in operation are British Magnox reactors built in the course of the sixties. The British electricity utility CEGB (Central Electricity Generating Board) operates 16 of these CO_2 -cooled reactors for electricity production at 8 sites. Important data for the Magnox stations are specified in Table I. The reactor power levels were continuously raised in the course of development while the efficiencies were improved by increasing the gas and steam temperatures and pressures.



Table I : Data of gas cooled reactors

Type	Station	Date of regular power operation	Number of reactors and steam generators		Power MW _e and MW _{th} 1) per reactor	
			R	SG	MW _e	MW _{th}
Early Magnox Reactors (steel vessel)	Berkeley	1962	2	16	138	558
	Bradwell	1962	2	12	150	531
	Hunterston A	1964	2	16	169	568
	Hinkley A	1965	2	12	250	971
	Trawsfynydd	1965	2	12	250	860
	Dungeness A	1965	2	8	275	840
	Sizewell	1965	2	8	290	948
Late Magnox Reactors (concrete vessel)	Oldbury	1968	2	8	300	982
	Wylfa	1971	2	8	530	1875
AGR	Hinkley B	1976	2	24	620	1500
HTR	AVR Jülich	1968	1	1	15	46
	Peach Bottom	1967 2)	1	1	40	115
	Fort St. Vrain	1976	1	12	330	842

1) Design values

2) Shut down Oct. 1974

The Table divides the Magnox reactors into early and late units. The late Magnox types (Oldbury, Wylfa) have prestressed concrete pressure vessel, in which the whole primary circuit is housed.

The Magnox series has been phased out with the Wylfa reactors. Successors are the similarly CO₂-cooled AGR reactors in which higher gas outlet temperatures are reached (above 600°C as compared to 350-400°C in the Magnox reactors). The first AGR reactors were completed and put into operation in the mid-seventies. The Table specifies the two Hinkley B AGR's for which data on the steam generator behaviour are available.

In the Federal Republic the AVR experimental power station is the only gas-cooled reactor in operation for which comparable operating experience has been obtained. The coolant gas used

is helium which reaches a temperature of 950°C at the core outlet at a pressure of about 10-11 bar.

In the USA the helium-cooled Peach Bottom reactor was in operation from 1967 to 1974 with a thermal power of 115 MW. Hot gas temperatures of 730°C were reached at a coolant gas pressure of 24 bar. Since 1976 the Fort St. Vrain reactor has been operated as a prototype plant. Its thermal power amounts to 842 MW. Coolant gas pressure and hot gas temperatures are 40 bar and 770°C.

2.1 Steam Generators

Three principles are usually applied for steam generation in the Magnox reactors /4/, /5/:

- natural circulation process
- forced circulation process
- once-through process.

The steam generators of the Magnox reactors were produced by four manufacturers providing a broad spectrum in terms of design and manufacture.

The early Magnox reactors work predominantly on the forced circulation principle according to which water is recirculated by pumps in the steam generator tubes and heated. The steam produced is separated in steam drums, superheated and then fed to the turbines /6/.

The two Dungeness A reactors produce steam according to the natural circulation process. This involves recirculation of the water in the tubes due to natural convection. The steam is separated in steam drums as in the case of forced circulation.

The late Magnox types and the succeeding AGR's have once-through steam generators as are also provided in high-temperature

reactors. The characteristic feature of the once-through system is the fact that all of the water entering the steam generators arrives at the steam generator outlet as superheated steam so that no steam drums are required.

It should also be mentioned here that two independent steam cycles of different pressure levels for high-pressure turbines and low-pressure turbines are integrated in the Magnox steam generators, except for Wylfa. Table II contains essential thermodynamic data.

Table II: Data of gas cooled reactor steam generators (SG), design values

Station	exchanged heat per SG	inlet gas temperature °C	outlet gas temperature °C	coolant pressure bar	HP-steam		LP-steam	
	MW				°C	bar	°C	bar
Berkeley	73	168	345	9	322	22	322	5
Bradwell	89	175	390	10	372	54	372	15
Hunterston A	71	205	380	10	374	39	374	10
Hinkley A	162	190	378	14	363	48	349	14
Trawsfynydd	143	184	392	18	375	67	365	22
Dungeness A	210	220	410	20	393	98	395	41
Sizewell	237	220	410	19	391	48	390	19
Oldbury	223	235	412	25	400	97	393	49
Wylfa	469	230	414	28	396	48	—	—
Hinkley B	125	285	650	41	540	160	541	41 ¹⁾
AVR Jülich	45	275	950	11	505	73	—	—
Peach Bottom	115	340	730	24	540	100	—	—
Fort St. Vrain	70	400	770	49	538	170	538	48 ¹⁾

¹⁾ Reheater

The steam generators of the early Magnox reactors are of the usual column design with round cross-section. (Typical dimensions: $\varnothing = 5-6$ m, height about 20-30 m). The heating surfaces are of the order of magnitude of 10,000-50,000 m².

In the late Magnox reactors the steam generators are accommodated in a prestressed concrete vessel together with the reactor

core (integrated construction). The two Oldbury reactors have six steam generators each in the usual column configuration, but with rectangular cross-section. In the case of the Wylfa reactors, the steam generator system is designed as an annulus around the reactor core. In the literature it is sometimes regarded as one steam generator (see /4/) and sometimes as four steam generators (4 x 90° segments, see /7/). The Wylfa steam generator design is unique. The failure balance of the steam generators shows that the compact construction aimed at in Wylfa presents no advantages over the earlier conventional steam generators.

The Hinkley B AGR steam generator system follows the Oldbury design. Twelve steam generator columns with rectangular cross-sections are arranged around the core in an integrated construction.

The steam generator of the AVR experimental power station is arranged directly above the reactor core /8/.

While it was still possible to use carbon steels or low-alloy steels for the Magnox reactors, the superheater tubes of the AGR's had to be manufactured from austenitic steels (steam conditions 550°C, 160 bar). A sensitive although technically controllable area is the connection (bimetal weld) between ferritic and austenitic steam generator tubes.

Although the AVR test reactor is the gas-cooled reactor with the highest coolant gas temperature (950°C) at the core outlet, temperature-resistant ferritic steels (10 CrMo 910) were still used for the hottest steam generator tubes, since a main steam temperature of 505°C and a pressure of 73 bar at the superheater outlet still permit the use of this material.

The American HTR's have been equipped with helically coiled, once-through steam generators. Similar to the AGR's it was

also necessary for these reactors to select high-temperature alloys for the tubing in the hottest sections. Carbon steels were partially used in the preheaters followed by sections made of 2 1/4 Cr - 1 Mo. INCOLOY-800 was used in the hottest superheater zones /12/.

3. STEAM GENERATOR FAILURES IN GAS-COOLED REACTORS UNTIL 1979

In the following, the failure data will be specified and the causes, sizes and progression in time of the failures will be discussed.

3.1 Total Number of Failures

A total of 313 leakages occurred at the reactor stations listed in Table III during roughly 1500 calendar years of steam generator operation; these were almost exclusively heating tube failures /3/.

This gives a failure frequency of 0.21 per steam generator and year for an average operating age of the steam generators of approx. 11 years (these are calendar years, downtimes and part-load operation are not included). The highest failure frequency is found for the Wylfa steam generators amounting to almost 0.7, whereas the AGR's have not yet exhibited any failures in the observation period. Among the steam generators in operation for a longer period of time, the 8 Dungeness A units have proved to be most reliable. Only a single failure occurred in more than 13 years of operation.

3.2 Causes and Sizes of Failures

Most of the steam generator failures in the reactors listed in Table III occurred in the first few years of operation and were identified as being caused by faulty design. Faulty design will be understood here as both defects in design

and systematic defects due to manufacture as well as undue deviations from the design conditions of operation (e.g. inappropriate water chemistry). Such systematic defects cause about 80 % of the failures (see /3/). The remaining failures can be regarded as random failures.

Table III

Station	Number of steam generators	Years of operation (until 1979)	Years of steam generator operation	tube failures
Berkeley	16	16.5	264	10
Bradwell	12	16.5	198	112 a)
Hinkley A	12	14	168	4
Trawsfynydd	12	14	168	39 b)
Dungeness A	8	13.5	108	1
Sizewell	8	13	104	16 c)
Oldbury	8	11.5	92	6
Wylfa	8	8.5	68	46 d)
Hunterston A	16	11 (until 1975)	176	75 e)
Hinkley B	24	3	72	0
AVR	1	10	10	1 f)
Peach Bottom	2	7.5	15	2
Fort St. Vrain	12	5 (until 1981)	60	1

1503

313

- a) mainly weld defects which led to leakages in the high-pressure section during the first few years of operation;
- b) most of the leakages caused by poor water quality (mainly high-pressure section);
- c) 10 tube failures in the 7th and 8th year of operation in the high-pressure section, 3 leakages after 1975 in the low-pressure section;
- d) corrosion, erosion (fretting), defects in design
- e) predominantly failures in the low-pressure section occurring in the first few years of operation; one large leak in the cm² range in the high-pressure superheater;
- f) small tolerable leaks on the tube plate /13/.

The failures at Bradwell, for instance, were chiefly caused by poor weld seams leading to water ingress after a short period of operation. The failures at Trawsfynydd resulted from corrosion effects due to inadequate water chemistry.

General causes of failure leading to steam generator leakage in gas-cooled reactors essentially include

- corrosion due to poor feedwater quality
- corrosion-erosion effects
- fatigue due to vibrations
- fretting due to vibrations.

The typical steam generator leak develops due to erosion of the water (or steam) passing from the secondary into the primary circuit from extremely small dimensions up to cross-sections which lead to significant increases in the moisture contents of the coolant gas. Such evolution processes can take days to months. The maximum leaks produced are several mm² in size.

Leak sizes in the cm² range are rare events. They occur as a result of massive tube wall thinning over larger areas. In tube bends, for example, the magnetite layer on the water side can be continuously destroyed (chipping, reconstruction, renewed chipping etc.) due to alternating local thermal stressing, which may lead to a high loss in wall thickness. For the reactors listed in Table III, only one leak of this size occurred at Hunterston A during the 1500 years of steam generator operation examined.

Major damage such as the simultaneous failure of several tubes - caused, for example, by the impact of broken pipes - has not occurred to date in the reactors specified. Damage such as the failure of tube plate connecting cylinders or thick-walled headers has not yet been experienced either.

3.3 Failures in Different Steam Generator Sections

The Magnox reactors (except for Wylfa) are provided with two steam cycles (high-pressure section, low-pressure section). From the respective start of operation until 1975 data are available on the distribution of steam generator failures between the two cycles /2/. The total of 251 tube failures is divided into 86 for the low-pressure section and 165 for the high-pressure section. As many as 90 high-pressure failures, mainly in the preheater region, were encountered at Bradwell, while Hunterston dominated with 55 leakages - mainly superheater - in the low-pressure section.

It has been reported in /3/ that the failure distribution over the steam generator sections preheater, evaporator and superheater during the period from 1975 to 1979 can be essentially regarded as uniform.

The AVR high-temperature test reactor had a small leak in the superheater section /9/. Another small leak in the superheater bundle from INCOLOY-800 occurred in the Fort St. Vrain HTR /10/.

3.4 Progression in Time of Failures

It has already been mentioned in Section 3.2 that most of the failures occurred in the first few years of operation. The following Table IV shows the number of failures for the individual years of operation over a period of 8 years for eight Magnox stations with a total of N = 92 steam generators.

The cumulated number of failures per steam generator and year decreases from 0.8 to about 0.3 which clearly indicates that defects in design lead to damage especially during the initial period of operation.

Table IV: Progression in time of failures for 8 Magnox stations with $N = 92$ steam generators over a period of 8 years

t (year)	failures in the t-th year	cumulated number of defects Σ	$\frac{\Sigma}{N \times t}$
1	75	75	0.815
2	64	139	0.755
3	41	180	0.652
4	13	193	0.524
5	19	212	0.461
6	3	215	0.389
7	6	221	0.343
8	9	230	0.313

When $\Sigma/(N \cdot t)$ is approximated by a function f of the form $f = A \cdot t^\beta$, $f = 0.96 \cdot t^{-0.49}$ (t in years) is obtained. The quantity $\Sigma/(N \cdot t)$ thus shows a more or less $1/\sqrt{t}$ behaviour; or rather, the cumulated number of defects ($\Sigma = N \cdot A \cdot t^{\beta+1}$) is about proportional to \sqrt{t} .

4. DERIVING A PROBABILITY FOR THE OCCURRENCE OF LEAKAGES AND A LEAK SIZE DISTRIBUTION

It can be seen from Table II that the steam generators are subjected to rather different loads. The steam generators of helium-cooled reactors are subjected to highest stressing. The question now arises of whether there is a correlation between stressing level, geometrical size and leakage probability.

A closer study of the failure data shows that the failure frequencies for the individual plants do not increase either with the geometrical size of the steam generators or with the thermal and pressure load. It can be seen, for example, that the steam generators with the largest heating surface, i.e. those of the Dungeness A plant, only exhibit one leakage during their entire period of operation. Moreover, not a single leakage failure has occurred in the AGR steam generators which are comparable to the HTR's with regard to temperatures

and pressures. The main factors of influence on the failure frequency are rather the design, manufacture, testing and the operating conditions.

It may be appropriate to critically view the failures that have occurred, before deriving an average failure frequency from the data material available. The large number of leakages in the Bradwell steam generators mainly results from systematic weld defects. The Wylfa steam generators exhibit an unusual design which, moreover, was modified to deviate from the original concept for want of space. Difficulties with the steam generators have been encountered from the onset of operation.

The failures at Wylfa and Bradwell will not be taken into consideration for deriving the probability of occurrence for failures; in this way, the progress and process of learning in steam generator development is accounted for.

Neglecting Bradwell and Wylfa leads to 155 failures in 1237 years of steam generator operation, resulting in an average number of 0.13 failures per steam generator and year (i.e. one failure per steam generator in about 7.5 years on average).

The heating tube failures are with one exception small leaks extending at most into the mm² range. Under the steam cycle conditions of a modern HTR concept, a leak cross-section of 1 mm² on the feedwater side would lead to a water ingress rate in the range of 0.1 kg/sec.

Heating tube failures in the cm² range are rare events. Only one leakage of this type has occurred in a bend of the evaporator section at Hunterston A. Related to 1237 years of steam generator operation, a value of $8 \cdot 10^{-4}$ per steam generator and year is thus obtained for leak cross-sections in the cm² range. The leakage rate under HTR steam conditions is in the range of 10 kg/sec for 1 cm² on the feedwater side.

Major damage such as the failure of a tube plate connecting cylinder has not yet been experienced. In the absence of empirical values, a figure of 10^{-6} per m and year as specified in WASH-1400 /11/ can be referred to for the failure of the connecting cylinder. The length of the cylinders is of the order of 1 m so that a failure probability of 10^{-6} per steam generator and year is obtained.

With a view to the effects of water ingress accidents the division according to leak sizes specified in the following appears meaningful. The probabilities of occurrence, W , specified for the individual categories are explained further below.

It should also be noted here that this is a probability of occurrence averaged in time for a steam generator operated for approximately ten years. It should be borne in mind that the failure probability at the start of operation is higher than after several years of operation.

1st category: small leaks in the range of a few mm²;
 $W_1 = 0.13$ per steam generator and year;

2nd category: medium leaks from a few mm² to a few cm²;
 $W_2 = 10^{-2}$ per steam generator and year
(geometric mean);

3rd category: large leaks greater than 2F with F the cross section of a boiler tube;
 $W_3 = 10^{-5}$ per steam generator and year
(geometric mean).

The majority of leakages falls under the 1st category so that the figure of 0.13 specified on page 220 is a measure of the frequency of small leaks.

The value available for medium leaks is at the upper category limit ($8 \cdot 10^{-4}$ for leaks in the cm² range). In order to account

for small leaks extending into the second category, W_2 is selected as the geometric mean of 0.13 and $8 \cdot 10^{-4}$.

Accordingly, the figure for large leaks results from the geometric mean of $8 \cdot 10^{-4}$ (leaks in the cm² range) and 10^{-6} (failure probability for tube plate connecting cylinders).

5. CONCLUDING REMARKS

The results obtained until 1980 for steam generator leakages give a good idea of the reliability of steam generators in gas-cooled reactors. It is certainly meaningful to update these statistics using above all the experience gathered with AGR's. Recent data on the reliability of AGR steam generators seem to indicate that the leakage frequencies will probably decrease if the latest data are included.

Technological progress in steam generator technology has thus made it possible to simultaneously increase the performance and reliability of steam generators in gas-cooled reactors.

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DYNAMIC SIMULATION OF STEAM GENERATOR FAILURES

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Abstract

A computer program will be described which is capable to simulate severe transients in a gas heated steam generator. Such transients may arise in the safety analysis of accidents resulting from failures in the heat removal system of an HTGR power plant.

Important failure modes which have to be considered are ruptures of one or more steam generator tubes leading to water or steam ejection into the primary system or anomalous operating conditions which may cause damage due to excessive thermal stress. Examples are the complete dryout as a consequence of feedwater interrupt in connection with continuing gas heating and the reflooding of the secondary channel with cold feedwater after dryout.

The steam generator program which is capable to simulate accidents of this type is written as a module which can be implemented into a program system for the simulation of the total heat rejection system. It based on an advanced mathematical model for the two phase flow taking deviations from thermal equilibrium into account. Mass, energy and momentum balances for the primary and secondary fluid and the heat diffusion equations for the heat exchanging wall form a system of coupled differential equations which is solved numerically by an algorithm which is stiffly stable and suppresses effectively oscillations of numerical origin.

Results of the simulation of transients of the type mentioned above will be presented and discussed.

1. Introduction

The computer program SIKADE-2, which is introduced with this paper is part of a program system which has been developed for the dynamic simulation of accidents in HTGR plants. It simulates the transient response of typical HTGR steam generators on failures in the steam generator itself or from abnormal operating conditions which may come about as a consequence of failures

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