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INADEQUATE ISOLATION OF CONTAINMENT OPENINGS AND PENETRATIONS

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
OECD NUCLEAR ENERGY AGENCY
38, boulevard Suchet, 75016 Paris, France

INADEQUATE ISOLATION OF CONTAINMENT OPENINGS AND PENETRATIONS

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Prepared by B. DE BOECK

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ABSTRACT

To mitigate the consequences of a core meltdown, a leaktight containment is very important. If the containment is to remain leaktight during the accident, it has to be leaktight at the onset of the accident. This report deals with the cases where the containment isolation function fails due to preexisting openings.

First the international experience is reviewed to assess the probability of preexisting openings. The conclusion here is that preexisting openings have a relatively high probability of occurence. It is therefore useful to be able to detect preexisting openings as soon as possible, ideally before an accident takes place, but also, as a back-up measure, in the frame of the accident management.

Methods to detect preexisting openings during reactor operation do exist. They are described and discussed in this report. Some countries have also implemented accident procedures to detect and isolate containment leaktightness defects after an accident has taken place. The U2 procedure in use in France is presented.

Finally the safety impact of preexisting openings is discussed. It is still difficult to quantify the risk reduction factor of containment leakage tests performed during reactor operation, but their potential to reduce the consequences of accidents and their usually low cost leads to the recommendation that their use be encouraged in all nuclear power plants.

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1. INTRODUCTION

The accidents at Three Mile Island and Chernobyl demonstrated the importance of containment integrity for mitigation of the risks of nuclear power plant operations. At a nuclear power plant, the containment is the final barrier to prevent the release of radionuclides to the environment during postulated accident situations. Because of its importance in mitigating the postulated consequences of an accident, it is necessary not only to assess its integrity after the accident has taken place, but also to ensure that it is leaktight before.

Typical allowable primary containment leakage rates are about 0.1% wt./day for PWRs and 1.0% wt./day for BWRs, but these rates are generally site dependent and may vary. Probabilistic risk assessments, including the Reactor Safety Study (WASH 1400), have indicated that containment leakage at, or slightly above design leakage may be a relatively minor contributor to overall nuclear reactor risk. The dominant containment related contributions to risk stem from accidents in which the containment ruptures or the containment isolation function fails or is bypassed. This report deals with the cases where the containment isolation function fails due to preexisting openings. International experience has shown that the probability of preexisting openings, as documented up to now in the literature, is relatively high. Therefore it is important to ensure, through appropriate means, that this probability is reduced as far as possible.

2. INTERNATIONAL EXPERIENCE

To mitigate the consequences of a core meltdown, a certain degree of leaktightness of the containment is very important. In normal operation, three barriers (the fuel rod cladding, the reactor coolant system pressure boundary, and the containment pressure boundary) protect the environment from the release of radioactive material present in the fuel. In many core meltdown accidents, the first two barriers are progressively breached and the containment represents the final barrier against releases. Therefore in severe accident analyses a great emphasis is placed on the containment behaviour [1], [8], [12], [13].

Of course if the containment is to remain leaktight during the accident, it has to be leaktight at the onset of the accident. Different studies have assessed the kind of preexisting openings and calculated the probability of containment function impairment.

Containment impairment probability is defined as the probability that the containment will not perform its function successfully at any given time during plant life. In this report, the containment is considered impaired when the leakage rate exceeds the limit of the Technical specifications.

In some reports the words "containment unavailability" are used instead of "containment impairment probability". This was not retained in this report because the containment can not be considered unavailable as soon as the leakage rate exceeds the limit of the Technical Specifications. Indeed, the containment leakage rate limits set in the Technical Specifications are clearly not set in direct relation to tolerable amounts of releases in the event of accidents, but primarily with regard to what can reasonably be achieved and can also provide a sound basis for surveillance and control of the leaktightness of the containment. Therefore, as will be shown in section 4, leakages in considerable excess of the Technical Specifications limit (up to a factor of ten, i.e. corresponding to a leak area of about 0.1 cm2, or 0.01 sq. inch) do indeed not contribute significantly to the risk of accidental releases.

Intolerable containment impairment can be due to excessive valves or penetrations leakage, valves or penetrations left open after testing or power transitions, holes drilled through the containment and left unsealed, airlocks failure. The leakage area can be very small or very large (from less than 1 cm2 to several square meters for a personnel airlock). The time it remains open may also vary widely. An airlock with both doors open will normally not remain so for long, but some leaks may go undetected until the next containment leakage test.

An extensive study of the reliability of the containment was performed in NUREG/CR-4220 [2]. Licensee Events Reports (LERs) and Integrated Leakage Rate Test (ILRT) reports were analysed to gather information about the containment performance. The containment impairment probability was then calculated from this information. A preliminary estimate obtained from the LER data base of containment impairment probability due to large leakage events (holes in the containment liner or open containment isolation valves) is in the range of .001 to .01. Containment impairment probability for relatively small leaks which violate plant Technical Specifications is estimated to be 0.3.

Estimates of the probability of a containment leak as a function of the leak area were obtained using ILRT data. A first analysis using only the ILRT reports gave the following results for PWR containment impairment probability versus leak area:

Impairment probability	Leak area sq. inches	(cm2)
0.022	0.001 to 0.01	(0.0065 to 0.065)
0.055	0.01 to 0.1	(0.065 to 0.65)
0.033	0.1 to 1	(0.65 to 6.5)
---------		
total 0.11		

Because local leakage rate tests are typically performed before an ILRT, the leakage rate noted in an ILRT is smaller than the actual case. Therefore an additional review of "as found" leakages from local leakage rate test reports was performed. The failures identified in this review were added to the previous results. The containment impairment probability for PWR then becomes:

Impairment probability	Leak area sq. inches	(cm2)
0.05	0.001 to 0.01	(0.0065 to 0.065)
0.125	0.01 to 0.1	(0.065 to 0.65)
0.075	0.1 to 1	(0.65 to 6.5)
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total 0.25		

The corresponding figures for BWR are:

Impairment probability	Leak area sq. inches	(cm2)
0.161	0.001 to 0.01	(0.0065 to 0.065)
0.143	0.01 to 0.1	(0.065 to 0.65)
0.036	0.1 to 1	(0.65 to 6.5)
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total 0.34		

The total impairment probability is thus very close to the one obtained from the LER data base (0.3). This shows that in case of an accident, the probability that the containment will leak in excess of the safety analysis value, is in the range of 25 to 35%.

The impairment probability of the containment is an input needed for level 2 and level 3 PRA's. The Reactor Safety Study (WASH 1400) used a probability of 0.002 for primary containment failure from inadequate isolation of containment openings and penetrations. The value used in NUREG-1150 was 0.005. This value was derived largely from operational data, the Seabrook PRA and "expert opinion". When compared with the results of NUREG/CR-4220, it is clear that these values are representative for large openings, giving substantially larger leakage rates than the Technical Specifications limits.

The American experience has also been recently assessed in [3]. This study is more qualitative than NUREG/CR-4220. It analyses the kind and magnitude of openings discovered in American containments from 1980 through 1986.

The results of NUREG/CR-4220 have been reviewed in NUREG-1273 [11]. It was concluded that the findings in NUREG/CR-4220 were simplistic and used very conservative assumptions. A more refined analysis was thus performed. The impairment probability of the containment was computed as a function of the leakage rate of the containment expressed in terms of La, the allowable leakage rate of the containment. The results are the following:

Estimated impairment probability

La	BWRs	PWRs
1 to 10	0.10	0.31
10 to 100	0.04	0.08
G.T. 100	0.01	0.07

Compared to the results of NUREG/CR-4220, one sees that the impairment probability of BWR containments is lower and for PWRs it is slightly higher. In both cases, a greater share of the impairment probability is due to small leakages.

A study of the containment impairment probability has also been performed in Belgium. LERs do not exist for Belgian plants, therefore only ILRT reports were used. Because local leakage rate test reports give only "as left" results, the ILRT test reports were used alone. Twelve ILRT reports were analysed, covering about 36 reactor years. Three tests showed a containment failure. These are:

- Doel 2, september 1986 : at 0.5 bar overpressure, an open valve was discovered on the personnel air lock and was closed.

- Tihange 2, july 1981 : at 2.7 bar overpressure the equipment hatch began to leak. This hatch is bolted from outside and is therefore not autoclave. It was discovered that the torque used for the bolts was too low.

- Tihange 3, july 1984 : at 0.4 bar overpressure on open line was discovered on the equipment hatch. This line had been used previously for some tests and had been inadvertantly left open.

Using the same methodology as in NUREG/CR-4220, one obtains a containment impairment probability of 0.13 which is very close to the value of 0.11 obtained in that NUREG.

Reference [4] also contains a study of preexisting openings in technical annex 9. The study was performed in Italy by ENEA/DISP and was based on data taken from American LERs and Nuclear Power Experience Books. Data from the Italian Operating Experience were also used. The containment impairment probability versus leak area obtained in this study is the following :

	Impairment probability	Leak area, cm2
	0.088	L.T. 0.03
	0.084	0.03 to 0.3
	0.076	0.3 to 3
	0.008	3 to 30
	0.008	G.T. 30
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total	0.26	

The total impairment probability is close to the value obtained in NUREG/CR-4220, even if one subtracts the contribution from leak areas lower than 0.03 cm2, which would not give a real containment impairment.

When comparing the impairment probability versus leak area from different studies, care should be taken of the formula used to compute the leak area from the mass flow rate. The Italian study uses a formula which give results 70% higher than those of the formula used in NUREG/CR-4220. The formula chosen depends on the assumptions about the leakage path. In most cases the characteristics of the leakage path are either not known or very complex. Therefore instead of speaking of leak area, one should speak of "equivalent" leak area, corresponding to specific assumptions.

The leaktightness capability of the Caorso containment has been reassessed recently. This study, based on the local leakage rate test results, shows recurrent problems with the leaktightness of some penetrations. To allow the detection of problems, it is necessary that not only "as left" results be reported, but also the "as found" leakage rates.

The results of the global and local leakage rate tests of the Spanish nuclear power plants have been compiled in 1989. The "as found" and the "as left" leakage rates have to be determined. The main objective of this requirement is to obtain a better appreciation of the containment leaktightness as it is during normal operation. The compilation showed that some tests failed, but no gross failure was detected.

The OECD member countries report the incidents in their nuclear power plants through the Incident Reporting System. A recent study [5] screened the IRS data base and analysed 67 events involving a loss of containment functions. These events were grouped in five categories and analysed to identify the causes and remedies to the problems. The conclusions of the report are qualitative and stress the importance of the experience feedback to avoid the repetition of similar events. It was not the aim of the report to quantify the probability of the various failure modes assessed.

3. DETECTION METHODS

As described in section 2, the operational experience has shown that the containment function can be impaired by various causes. The present inspection methods and programs are not able to ensure that the containment will always perform as expected in accident conditions. In several instances important leakage paths have been found in the containment. Even if it is not always possible to avoid them, it is important to detect them as soon as possible.

Different detection methods, capable to monitor the leak-tightness of the containment during reactor operation, are possible. In subatmospheric and inerted containments, large leakages will be rapidly detected by the pressure or oxygen concentration increase. In Sweden the BWRs are kept at an underpressure of 1 to 10 kPa, and the PWRs at an overpressure of 10 to 20 kPa. This also allows the detection of large leakages. Other methods are described in NUREG-1273 [11]. The method which is now routinely used in Belgium and in France for PWRs with a large dry containment is described below.

In normal operation, the pressure in the reactor building has a tendency to increase owing to the leakage of the compressed air system. If one measures the flow of incoming air, the pressure, the temperature and the humidity in the building, it is possible to calculate the leakage rate [6]. The absolute method and/or the reference vessel method can be used.

For a typical test, the pressure is allowed to go from -20 mbar to +60 mbar. At least the range between 0 and 50 mbar should be covered if one wants to have a reasonable accuracy. The pressure increase rate is normally in the range of 0.5 to 1 mbar/h. A test therefore lasts for a few days.

The minimum test duration should be 50 hours, to be able to get enough data points. If during the test, the atmospheric pressure drops suddenly and the maximum differential pressure is reached before 50 hours, the test should be performed again.

All the parameters are measured every 30 seconds. The values are averaged over 15 minutes and this gives one data point. A typical test gathers 200 to 400 data points. This is done by means of a personal computer. The data points are plotted in a graph showing the leakage rate as a function of the square root of the differential pressure between the reactor building and the auxiliary building.

During the test, care should be taken not to disturb the conditions in the reactor building. Airlocks movements should be avoided as far as possible. The ventilation and the cooling of the containment should be very stable. The method takes care of temperature variations but the representativity of the temperature measurements is never perfect. Therefore any disturbancies in the temperature distribution in the containment will lead to a greater spreading of the data points.

The absorption of air in the concrete of the internal structures of the containment is also a potential source of errors. When the pressure in the containment increases, some air goes in the concrete and the apparent leakage rate of the containment is greater than the real value. The opposite is true when the containment is depressurized. This is one of the reasons why a stabilization period is required during the integrated leakage tests performed at an elevated pressure. An experimental study performed in Belgium has shown that the time constant of this phenomenon was about 14 hours at Doel 3 and 7 hours at Doel 4. The value is plant specific because it depends on the quality of the paint covering the concrete, but it is believed that these values can be considered as a typical range. Therefore, for the tests performed during reactor operation, and in view of the long duration of these tests (at least two days), the low pressure, and the relatively large leakage rates that one wants to detect, the effect of air absorption in concrete is considered to be negligible.

The tests performed in Belgium use the same instrumentation as the ILRT's, with the addition of the flow-meters on the compressed air system. These are thermal flow-meters which need no correction for temperature and pressure. To save a penetration, the pressure difference between the containment and the auxiliary building is not measured directly, but is computed from absolute pressure measurements. The absolute pressure in the containment is also needed to compute the mass of air.

The temperature is measured by about 30 platinum probes distributed in the containment volume so as to give the best possible average temperature. The humidity is measured by 5 to 10 lithium chloride probes. In the absolute method the air mass change in the containment during each time step is computed from the absolute pressure, the temperature and the humidity. In the reference vessel method the air mass change is computed from the absolute pressure, the pressure difference between the reference vessel and the containment, and the humidity. For both methods, the free volume of the containment must be known.

The difference between the air mass change computed from the parameters in the containment, and the air mass change measured by the flow-meters on the compressed air system, is the leakage flow of the containment. This leakage is then plotted versus the square root of the differential pressure between the reactor building and the auxiliary building.

A straight line is then computed by the least squares method. Conventionnally the leakage rate is expressed as the difference between the value at 60 mbar and the value at 0 mbar, and is noted Qf60. The value at 0 mbar (noted Qf0) should theoretically be zero, but is nearly never so, for two reasons :

- the systematic errors of the instrumentation and the error on the free volume of the containment
- an unaccounted inflow or outflow of gas, which is independent of the pressure in the containment.

The results of some of the tests performed in Belgium are given in [7]. These tests have shown that when the containment is leaktight, Qf60 is lower than 5 Nm3/h. Therefore if one measures a value higher than 10 Nm3/h, one is nearly sure that there is a leak. A leak of 10 Nm3/h at 60 mbar corresponds to a hole of 0.7 cm in diameter. Such a hole cannot yet be called a containment failure because it corresponds to a leakage rate of about ten times the Technical Specifications limit.

The leakage rate tests performed during reactor operation detect only the openings which are open to the containment atmosphere. Leaktightness problems in systems which are closed during normal operation but which could be open to the containment atmosphere after an accident are of course not detected by those tests. One important example is the main steam isolation valves in BWR. Numerous problems have been reported with these valves [5], but they can be detected only during local tests. Because a global test performed during operation cannot test all penetrations, it is not a substitute for the local tests.

Different alternate leakage testing methods were analysed in NUREG-1273 [11]. The conclusions of the analysis are the following:

(1) Although some alternate methods of checking containment isolation integrity appear practical and sufficiently sensitive, these methods do not have the accuracy of the normal ILRT's. However, these methods seem to offer enough accuracy and speed of detection to justify their use for detecting gross leakages.

(2) The current containment integrity testing program is capable of detecting all reported events documented in the LER database in NUREG/CR-4220. Nevertheless, the alternate test methods offer one advantage over current testing techniques. This advantage is speed of detection, which can range from one day to several weeks, whereas the leaks detected by the conventional methods could have existed for an average of 6 to 12 months before detection.

(3) The alternate test methods should not be considered a complete replacement for normal ILRT's, because all of the alternate methods are intended for use at reduced pressure under standard operating conditions. Thus, these methods do not test plant equipment under higher containment pressure.

The same conclusions were reached in Belgium, and confirmed after the method described above had been used for some time, with one addition: because the alternate method covers part of the functions of ILRT's, the frequency of the ILRT's can be reduced to once every ten years [6].

A complementary measure to reduce the consequences of containment isolation failures (either due to preexisting openings or a failure of the containment isolation system) is to prepare an accident procedure to help in the detection and the isolation of an opening, after the accident has taken place. Of course priority should be given to the prevention of containment isolation failures, but, in line with the defence in depth concept, means should be available to the operator to take corrective actions after the onset of an accident. Such a procedure exists already in France (procedure U2) and maybe in other countries.

The U2 procedure [14] addresses the search for and processing of abnormal containment leaktightness defects after an accident has caused a fuel degradation and/or a primary system defect. The U2 procedure must in fact cover a wide range of accident severity, because it is obviously desirable to activate it as soon as any threat of significant release of radioactivity inside the containment has been discovered. It defines:

- the condition of containment surveillance (radioactivity at the stack, in the sumps and inside the containment, state of containment isolation systems),

- the actions to be taken to mitigate the radioactive releases (for example: isolation of a unit, reinjection of liquid waste inside the reactor building).

Currently, four distinct actions to recover or improve the radioactivity confinement are operational on 900 MWe units, which are actuated during the SPI or SPU procedures whenever activity is detected at the stack, in the containment, in the reactor coolant system or above the sumps of the nuclear auxiliary and fuel buildings (Figure 1).

If a significant radioactivity level is detected at the stack (greater than Level 1 or greater than level 2 during a time lapse T1), actions aim at isolating the contaminated areas and at putting an end, when feasible, to the leak, once its origin has been detected by an instrumentation channel devised for the purpose; such a channel is dedicated to provide the activity flow rate at specific places in the ventilation ducts of the Nuclear Steam Supply System.

If a very high level of activity is detected in the containment (dose rate greater than Level 3) - which means a failure of the two first barriers, i.e. the fuel cladding and the primary system boundary, - all the containment penetrations are checked, and identified failures, if any, are repaired so as to limit radioactivity releases into the environment.

If a high level of activity is detected in the primary coolant (dose rate greater than Level 4), the penetrations in contact with the highly contaminated primary water are isolated except for those of the safeguard systems, because the former circuits, external to the containment building, are not dimensioned for a degraded fuel accident; the reactor is then brought to the depressurized cold shutdown state. Nevertheless, there is a possibility of using the Nuclear Sampling System to monitor the primary coolant activity by making local adaptations to ensure the reinjection of the samples into the containment after analysis.

If a high level of activity (dose rate greater than Level 5) is detected above the sumps of the auxiliary buildings, there is a tightness defect in a safeguard system; in this case, an intervention team tries locally to put an end to the leak, when feasable. Injection to the Liquid Waste Treatment System is closed and the operator may reinject the sump water into the reactor building.

4. IMPACT ON SEVERE ACCIDENTS RISK

To quantify the safety benefit of performing leakage tests during reactor operation, it is necessary to calculate the risk of preexisting openings, and to see which part of this risk can be reduced by the tests. The ideal starting point would be a level 3 PSA where the results are given for different containment leakage rates and their associated probability. Such a study does not seem to exist up to now.

NUREG-1150 takes preexisting openings into account, but in a simplified way, and, from the main report, it is not possible to extract their impact on the results.

A specific study was performed in 1983 at Oak Ridge [9]. In this study, the risk for light-water reactor accidents has been evaluated as a function of containment building leakage rates. The analysis used the set of generic source terms and frequencies of occurences developed as representative of the range of postulated type of accidents applied at that time in reactor safety research.

The conclusion of the study was that the LWR accident risk is relatively insensitive to the containment building leakage rate. A large increase of the containment leakage rate causes only a small increase of the risk. A containment building leakage rate of 100%/day instead of 0.1%/day gives a 15% increase in the overall risk. This conclusion is mainly due to the fact that for the accidents and associated source terms frequencies chosen, the risk is dominated by severe core damage accidents with early containment failure. Such an assumption leads necessarily to the conclusion that the leakage rate of the containment is of minor importance. For the same reason, the same conclusion was reached in NUREG-1273 [11].

Since 1983, a lot of research has been performed in the area of severe accidents. Analytical tools have been improved and accident management studies have shown a large potential for reducing the probability of early containment failure. The NUREG-1150 has also shown that the risk and the relative contributors to the risk are plant specific. Therefore the quantitative impact of preexisting openings on the risk is plant specific and the conclusions of NUREG/CR-3539 [9] are not necessarily applicable to all LWR's.

At Oskarshamn 1 for example, OKG has performed an analysis with MAAP 3.0, of a postulated core melt accident with a small diameter (15 mm) leakage path from the containment to the environment [10]. The analysis clearly shows the importance of containment isolation and also the benefit of containment spray operation when it comes to the external source term. The cesium release was calculated to be 0.22% of the core inventory if the spray was started after 8 hours, and 0.04% if the spray was started after 33 minutes.

Inadvertant openings result in a leakage path that will probably end in the auxiliary building. One important aspect is the issue of habitability. In the case of failure of the emergency ventilation in the auxiliary building the access to vital areas might be prohibited due to high radiation dose rates. Calculations have been performed for the total blackout sequence at the Forsmark reactors in Sweden. The results obtained implied severe habitability restrictions to vital areas within 24 hours after the accident. This could make recovery actions much more difficult, but this kind of impact is not quantified in risk analyses. Post accident decontamination, on- and off-site, would also be more complicated in case of increased containment building leakage, at least for accidents not leading to containment failure.

Another problem is related to a potential threat to the containment structural integrity. If enough air is pushed through the opening in the first phase of the accident and if the steam is rapidely condensed later (e.g. by spray actuation), a dangerous subatmospheric condition could be created in the containment. The problem applies to mid-size openings, because for small openings the amount of lost air is small, and for large openings enough air can be sucked back in the containment.

For all these reasons, containment building leaktightness has to be assured as far as possible, regardless of the calculated impact on risk.

5. CONCLUSIONS

At a nuclear power plant, the containment is the final barrier to prevent the release of radionuclides to the environment during postulated accident situations. Because of its importance in mitigating the postulated consequences of an accident, it is necessary not only to assess its integrity after the accident has taken place, but also to ensure that it is leaktight before.

The operational record of nuclear power plant operation has shown several instances of failure of the containment isolation function. The probability that the containment leakage rate exceeds the allowable limit at any given time during plant life has been assessed in section 2 and has been shown to be relatively high. In most of the cases the leakage rate was not much higher than the allowable limit, but in some cases it was such that the potential environmental consequences of an accident would have been large.

The containment leakage rate limits set in the Technical Specifications are clearly not set in direct relation to tolerable amounts of releases in the event of accidents, but primarily with regard to what can reasonably be achieved and can also provide a sound basis for surveillance and control of the leaktightness of the containment. From section 4 it is thus clear that leakages in considerable excess, e.g. by a factor of ten, of the Technical Specifications limits do indeed not contribute significantly to the risk of accidental releases.

To study the reliability of the containment isolation function, it is important that the local and integrated leakage rate test reports record all findings, i. e. "as found" leakage rate, corrective measures if any, and "as left" leakage rate. This information is also important to allow the detection of recurrent problems with some penetrations.

Probabilistic risk assessments have indicated that large preexisting openings is a potential contributor to overall nuclear reactor risk. The determination of the impact of preexisting openings on accident risks is still affected by the large uncertainties on the probability of early containment failure. If the risk is dominated by early containment failure, the containment leaktightness is not very important. At present, efforts are under way to reduce the probability of early containment failure. This increases the importance of the containment leaktightness and it is therefore important to avoid preexisting openings or to detect them as soon as possible.

Local leakage rate tests performed at each refueling are one method to achieve this, but one is never sure to cover all possible leakage paths. The global containment leakage rate tests are able to accurately detect all leakage paths but these tests are time consuming and are therefore performed only one to three times every ten years. Openings in the containment could go undetected for several years.

In section 3 it has been shown that methods exist to detect large openings in the containment during reactor operation. These methods do not have the same accuracy as the pressure tests of the containment (ILRT's), but they can be performed continuously and are able to detect leaks in the containment that are smaller than the ones which would be a concern for accident conditions. Methods to assess the leaktightness of the containment during reactor operation are used routinely in some countries.

In section 3 it has also been shown that accident management measures exist to detect and isolate openings in the containment after an accident has taken place. Of course priority should be given to the prevention of containment isolation failures, but, in line with the defence in depth concept, means should be available to the operator to take corrective actions after the onset of an accident. This is already the case in some countries.

6. RECOMMENDATIONS

The safety benefits to be gained from global reactor containment leakage rate tests performed during reactor operation are still difficult to quantify. To perform this quantification, it is necessary to calculate the risk of preexisting openings, and to see which part of this risk can be reduced by the tests.

Even if it is difficult to quantify the severe accident risk reduction offered by such tests, they have the potential to reduce the consequences of accidents and their cost is usually low. It is therefore recommended that the use of such tests be encouraged in all nuclear power plants, taking into account the design and operational characteristics.

The possibility to detect and isolate inadvertant openings in the containment after an accident has taken place has been studied in some countries. This has led to the implementation of accident procedures to cover this specific aspect. It is recommended to study such accident management measures and to assess their applicability to all nuclear power plants.

Given that preexisting openings have been shown not to have a negligible probability, and given the uncertainties still affecting their impact on overall risk, it is recommended that a study be performed of the various PSA available to compare how preexisting openings were taken into account and what their impact is on the results, and it is also recommended that specific calculations be performed of the amounts of radioactivity released as a function of leak area.

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U2 PROCEDURE

(Four maneuver sheets)

Figure 1