Studsvik Report

STUDSVIK/K4–79/28

CRITERIA FOR RECLADDING OF SPENT LIGHT WATER REACTOR FUEL BEFORE LONG TERM POOL STORAGE

SKI B56/78, Final report

Kjell Pettersson
Lennart Jansson

STUDSVIK ENERGITEKNIK AB

STUDSVIK/K4-79/28

1979-06-19

Kjell Pettersson
Lennart Jansson

Statens Kärnkraftinspektion
SKI B56/78 Final Report

CRITERIA FOR RECLADDING OF SPENT LIGHT WATER REACTOR FUEL BEFORE LONG TERM POOL STORAGE

SUMMARY

The question of the need for any special treatment of failed fuel elements prior to long term pool storage has been studied. It is concluded that the main problem appears to be hydride embrittlement of failed fuel rods, which may lead to increased damage during handling and transport of the failed fuel. Some mechanisms for the degradation of failed fuel rods have been identified. They can all be considered as relatively improbable, but further experimental evidence is needed before it can be concluded that these degradation mechanisms are insignificant during pool storage. The report also contains a review of methods for identification of leaking fuel bundles and fuel rods.

Godkänd av: RB Forsyth

1 INTRODUCTION

Due partly to the slow build-up of reprocessing capacity for light water reactor fuel, it will be necessary to store the fuel elements for periods of 20 - 50 years before reprocessing the fuel, or any other means for final disposal of the fuel decided upon. In a typical facility for interim storage, the fuel elements will be stored in water pools where the temperature will be around 30°C. Before interim storage the fuel elements will have been stored for at least a year in the spent fuel storage pool at the reactor site.

In the present work, which has been sponsored by the Swedish Nuclear Power Inspectorate, one question in relation to pool storage of light water reactor fuel elements has been studied: Should defective fuel elements be subjected to any special treatment prior to or during interim storage?

The safety of long term storage of spent fuel is based on the expectation, that there is no degradation mechanism for the Zircaloy cladding which will threaten the mechanical integrity of the fuel elements during the time spent in pool storage. This question has been the subject of two recent reports by Johnson (1) and Vesterlund and Olsson (2), where a number of possible degradation mechanisms have been analysed and available spent fuel storage experience has been summarized. The conclusions of both Johnson and Vesterlund/Olsson are that there are no currently known degradation mechanisms which limit the storage times to less than 50 - 100 years, provided that the fuel is stored in pure water, and that direct contact between aluminium alloys and the Zircaloy cladding is avoided. However Johnson recommends exploratory examinations of selected pool-stored fuel to define whether any slow degradation mechanism of the fuel and bundle materials is operative.

The subject of the present report is limited to the question of the handling of already defected fuel rods or fuel elements. Should they be subjected to any special treatment prior to long term pool storage, like recladding of failed fuel rods, or should elements with failed fuel rods be stored in special containers? And where should the recladding or placement in a special container take place: at the reactor pool, or at the reception pool in the storage facility? The question of the need for any special treatment of failed fuel has also been analysed by Vesterlund/Olsson (2), and by Johnson in a later report on the impacts of reactor induced cladding defects on spent fuel storage (3). Vesterlund/Olsson analysed the possibility of special degradation effects for defected fuel, and concluded that any such effects were unlikely to occur with rates significant for pool storage. However they did not seem 100 % certain that internal hydriding could be excluded during pool storage; they also cited a few cases of storage of spent fuel which confirmed their conclusion. The same cases were reported by Johnson, together with a few more cases of storage of defective spent fuel with reactor-induced defects (3). His report contains no discussion of special degradation mechanisms for failed fuel.

2 POTENTIAL PROBLEMS FOR THE STORAGE OF DEFECTIVE FUEL

The following problems for the storage and handling of defictive fuel were considered relevant, and are treated in detail in later sections:

a) Release of fission products to the pool water. Relationship between defect size and fission product release. Characterization of defects.

b) Handling and transport accidents due to brittleness of failed fuel pins. Determination of the amount of hydriding.

c) Degradation mechanisms for defective fuel rods during long term storage (with the exclusion of all degradation mechanisms which are common to intact and defected rods).

All these problems are interrelated, and in that respect this particular subdivision is arbitrary. Our work should be regarded as a rather general discussion by fuel specialists, of what can be done to minimize problems with storage of spent defected fuel. We have no competence to discuss the risks involved in storage and handling of defective fuel, but can only discuss what could possibly happen to the fuel under certain circumstances, and also to some extent which fission products could be released.

3 EXPERIENCE WITH HANDLING AND STORAGE OF DEFECTIVE FUEL

The experience of handling and storage of defective fuel rods, reported by Johnson (1,3) and Vesterlund/Olsson (2), can be described as remarkably free of large problems. Johnson reports no handling accidents with defective bundles. Of 9 reported handling accidents with intact bundles, only one resulted in a detectable radiation release. For the problem concerning handling accidents with defected fuel rods, it is interesting to note that in one case when a bundle dropped 1.5 m some of the rods were permanently bowed after the incident. Since it is likely (as will be discussed later) that a defective rod will be hydrided, it seems possible that dropping a bundle containing defective rods would have resulted in severe damage to the defective rods.

Relatively minor problems with releases of radioactivity during the storage of defective fuel rods are reported (1,2,3). No visible degradation of defective rods is reported. Whereever possible we have examined the sources used by Johnson and Vesterlund/Olsson, and their accounts must be regarded as accurate. From Mol Hagsgård (4) reports problems with fission product release from defective fuel rods in connection with work in the reception pool. Both gaseous and water soluble fission products were released. The fuel was received as early as three months after irradiation in the reactor, and therefore the experience with releases of gaseous fission products may not be relevant to the present problem, since after one year an additional 30 and 50 half-lives of the most important gaseous fission products, I-131 and Xe-133 respectively, will have expired. What is relevant, however, is the observation that the defective fuel is much more damaged after transport, than it was reported to have been before loading into the transport container. Hagsgård describes the problem as "transport damage due to brittle cladding and widening of old cracks".

This description suggests that the problem may well be caused by hydriding of defective fuel rods during operation in the reactor, which causes brittleness of the cladding at lower temperatures (see section 5).

For storage in the GE Morris pools, there is a requirement that no fuel may be transported to the pool if there is any visible loss of cladding or damage which might expose bare fuel pellets to the pool water (5). During the last several years of experience there has been no observed relationship between the quantity of fuel and the release of radioactivity to the pool water in the G E Morris pools. The variations in release rate have been associated with specific fuel receiving campaigns, and there is also some evidence that the release rates may be greater at higher water temperatures. Unfortunately, the report gives no values for the temperatures, only that with one of the three cooling units in use the temperature can be kept below 38°C.

At Windscale there is a requirement that reactor operators must report any leaking fuel, or fuel suspected of leaking before it is received by British Nuclear Fuels Limited.

From WAK (Wiederaufarbeitungsanlage Karlsruhe) in Germany, where defective fuel elements are stored in special canisters with loose-fitting lids, no problems either with handling or storage of spent fuel elements are reported (7).* Other German experience of spent fuel storage is similar (8). With regard to corrosion, Peehs et al state that under the controlled conditions of spent fuel storage, corrosion damage is completely out of the question.

---

* Some Cs-137 is released to the pool water from the seemingly intact fuel rods. It is unlikely that the Cs diffuses through the Zircaloy cladding, since from basic physical principles, it can be expected that Cs diffusion in a Zircaloy matrix is extremely slow at pool temperatures. More likely explanations to the Cs-137 releases are, either that there are undetected leaks among the intact fuel rods, or that the Cs comes from UO_2 contamination on the outside of the fuel pins (9).

The problem-free experience of storage of defective fuel, suggests that the main problem with defective fuel may be the increase in damage during handling and transport. However, we have not been able to find any reports concerning damage to fuel during transport. Reports on transport experience are mainly concerned with the safety to the public. In that respect there is to date an excellent record of transport experience with no activity releases at all to the environment (10, 11, 12).

4 IDENTIFICATION OF LEAKING FUEL BUNDLES

Several methods are used for the identification of leaking fuel bundles in light water reactors. The two oldest methods are wet sipping and dry sipping. The principle for these sipping tests is to achieve a temperature increase in the element which leads to an activity release if the fuel element is defective. More recently another method, vacuum sipping, has been developed.

4.1 Wet sipping (13, 14, 15, 16)

Wet sipping is based on the leaching of fission products, usually I-131 and Cs-134, from the defective element to a surrounding water volume. The greatest advantage with the method in BWR:s is that it can be used without removing the element from the core of the reactor. This is possible because the rods in a BWR-element are enclosed in a fuel channel which is open at both ends. Therefore each element is partially isolated from its environment. This isolation leads to a decrease in the heat transfer by convection to the environment, which results in a temperature increase for the fuel rods and a release of radioactivity. By placing a loose-fitting lid on the fuel channel the heat transfer can be decreased more, resulting in a further temperature rise. After a suitably long period of time (about 0.5 h), water samples are taken from the channel and are analysed for the fission products of interest. By comparing the radioactivity from the tested fuel channel with the average radioactivity of the reactor water it is possible to judge whether or not the fuel element is leaking. The sharpness of the test is reduced by convective water flow through the open lower end of the fuel element, and by unavoidable cross contamination between elements.

The PWR-elements have no fuel channels, and therefore it is necessary to remove the elements from the core before testing. When such an element is tested it is placed in a

closed water filled container for sipping. It has been clearly demonstrated that the sensitivity and reliability of wet sipping is increased when the test is performed in a completely separate volume of water.

The greatest disadvantage with wet sipping is that the concentration of suitable, easily leachable fission products is rapidly reduced by radioactive decay. Therefore wet sipping must be performed within a few weeks of the shut-down of the reactor.

4.2 Dry sipping

Dry sipping (17) works through the release of fission gas due to a temperature/pressure increase in the fuel rod. The fuel element is removed from the core and placed in a water filled container the lower end of which is open. The water in the container is then removed by filling it with air. This results in a temperature increase of the fuel element when fission gases are released to the air in the container. At a certain temperature level the container is again filled with water, and the air/fission gas mixture is removed and analysed.

An advantage with dry sipping is that the sensitivity of the test is much greater than for wet sipping.Due to the long half life of the isotope Kr-85 it is also possible to use dry sipping a long time after shut-down of the reactor. One disadvantage of the method is that it is possible to overheat the cladding if a mistake is made during the sipping procedure.

4.3 Vacuum sipping

In vacuum sipping the advantages of the two previous methods have been combined. Vacuum sipping gives the high signal levels typical of dry sipping, together with the inherent safety of the wet sipping technique, because the fuel element is surrounded by water during the test.

The element to be sipped is placed in a water filled container. The container may be located in the reactor vessel or in the spent fuel pool. The sipping container is connected to an evacuation tube at the upper end and an air tube with a gas sparger at the lower end. Air is supplied through the sparger, which serves to distribute the air as a mass of bubbles. The air and possible fission gases are trapped in an air pocket above the element. By bubbling the air through the container the pressure is reduced and the air sweeps up any fission gases which may have leaked from defective fuel rods. The activity of the air/fission gas mixture can be measured with a suitable detector. In a second stage, the pressure in the container is reduced and the differential pressure between the fuel rod and water serves to extract fission gases from defective rods. In a third stage, the partial vacuum is retained while the air in the air pocket is recirculated to sweep up any remaining fission gases released from the element. The time for vacuum sipping is less than 10 minutes, compared with 30 to 60 minutes for other methods (18).

4.4 Reliability of sipping results

The reliability of wet sipping tests was investigated for fuel elements from the Gundremmingen reactor (19). Fuel elements identified as defective by wet sipping were subjected to a more thorough investigation in the spent fuel pool, followed by a statistical evaluation of the relationship between the magnitude of the sipping signal and the presence of defects, as determined by the pool examination. The sipping signal (S) was in this case the ratio between the I-131 activity of the water in the shipping container and the average I-131 activity in the coolant water. The results are summarized below:

Table 1

S ≥ 100	At least one defective fuel rod with statistical certainty
100 > S ≥ 10	Presence of defective fuel rod/rods in 67 % of the cases
10 > S > 2	Presence of defective fuel rod/rods in 53 % of the cases
2 > S	Presence of defective fuel rod/rods in 25 % of the cases

The following table is also of interest:

Table 2

Sipping signal	Number of fuel elements with sipping indication	Number of defective rods	Fraction of defective rods
S < 2	299	33	0,3
2 ≤ S < 10	34	28	2,3
10 ≤ S < 100	21	37	4,9
S > 100	14	37	7,4

These results seem to indicate that there are many more suspected leakers than actual leakers. However, it should be remembered that these results cannot be used as a measure of the reliability of the wet sipping test for finding all leakers. There is no corresponding detailed examination of fuel elements with no sipping indication.

The high number of suspected leakers compared to actual leakers was explained as being caused by cross-contamination between elements. The wet sipping technique has subsequently been improved and according to ref.(15) the number of suspected leakers has been greatly reduced. There is however no quantitative qualification of the statement.

The vacuum sipping had, at the time of writing ref. (18), been used at five different BWR:s. In the table below the results are summarized.

Table 3

Site	Start date	Bundles sipped	Bundles resipped	Leakers identified	Approximate % of nonrecoil off gas removed
1. Peach Bottom 2	May 1977	426	2	12	95
2. Brunswick 2	September 1977	451	6	9	99
3. Hatch 1	March 1978	533	9	4	99
4. Peach Bottom 3	April 1978	339*	8	3	--
5. Browns Ferry 2	April 1978	764	4	3	--

*189 8 x 8 fuel elements were not sipped.

The removal of non-recoil off gas is said to be a measure of the sipping effectiveness. However, just as in the case of wet sipping there is no information available on the number of defective rods not identified by vacuum sipping.

In summary, techniques used for finding fuel elements with defective rods are based on the measurement of release of fission products. It is quite conceivable, and has been observed on test rods in the R2 reactor, that a defect may be healed so that no release of fission products occurs(20). It seems unlikely that such a rod would be detected in a sipping test, but it is quite possible that it might be hydrided and therefore brittle. There is at present no method available for identifying bundles with this type of rods, except the cumbersome method of examining elements on a rod by rod basis. Therefore it would be interesting to collect evidence on the hydriding of non-leaking defective fuel rods. The proposed PFFP program (9) may give relevant information on this problem.

4.5 Identification of defective fuel rods

When a fuel element has been identified as containing one or more defective rods, the problem remaining is to determine exactly which rods are defective. If a fuel element contains leakers, current practice seems to be to repair the element by exchanging the defective rods with new intact rods. Vesterlund/Olsson have given an impressive description of how this can be and has been done with ASEA-ATOM BWR fuel (2). The defective rods are identified by eddy current testing or ultrasonic testing. According to a specialist on eddy current testing (21) the method gives indications for both cracks and hydrides. Therefore it is possible to identify a failed rod if it is hydrided, even if the crack has closed so that the crack itself gives no indication in the eddy current test. With some development of the method it will also be possible to determine the amount of hydriding. Before eddy current testing, the rods must be removed from the element.

For PWR fuel made by KWU another method exists, which does not require removal of the fuel rods from the element, for the identification of a leaking rod (16). The fission gas plenum is heated electrically, which leads to vaporization of the water in the defective rod. The end plug is cooled so that the water vapour condenses there. The condensed water can be detected ultrasonically. The success of the method has been quoted as a stimulus for further development for use in BWR fuel repairs as well since considerable time saving could be achieved (22).

PWR fuel made by Westinghaus cannot be repaired as easily as KWU fuel. However it is possible to develop equipment so that individual rods can be withdrawn and tested, and if necessary be replaced by intact rods (23).

4.6 Relationship between sipping test results and defect size

No information has been found concerning this point. However it seems reasonable that when only a few defects are present it may be practical to identify individual rods. The classification of the defect should then be relatively simple, with exception of the determination of the degree of hydriding, which may require some development of eddy current testing (21).

5 HYDRIDING OF DEFECTIVE FUEL RODS

When a through-the-wall defect forms on a fuel rod it can be expected that some water enters the fuel rod. The fact that some defects subsequently close so that no fission products are released does not preclude that water has entered the fuel rod. It has been observed, for instance, that a defect opened under high stress in the cladding during a closed end burst test of a cladding tube, may close when the stress is released, and not open until the internal stress again reaches a high value (24).

The water entering the defect may cause hydriding of the cladding from the inside. The causes and consequences of internal hydriding of defective Zircaloy fuel have been the subjects of three reviews by Locke (25, 26, 27). According to Locke (26) hydrogen can be released by the following mechanisms:

1 Production of hydrogen by cladding corrosion.

2 Production of hydrogen by steam-UO_2 reaction.

3 Dissolved hydrogen in the coolant.

4 Radiolysis of the coolant entering the defect.

It is not clear which of these mechanisms is most important. The released hydrogen leads to internal hydriding of the fuel rod. The attack is often localized and so called sunbursts are formed. The localization of the attack is influenced by a number of factors such as:

- fission products which may influence cladding oxidation and therefore hydriding
- mechanical defectsin the internal oxide layer
- variation in the moisture content of the hydrogen along the fuel rod
- variation of the surface heat flux along the fuel rod.

Once a local hydride attack has started, the attacked spot serves as a "sink" for hydrogen and rapid hydrogenation can occur. Locke has analysed data from operation with defective fuel, and found a correlation between surface heat flux and days to failure after defection, Fig. 1. Failure is in this case defined as gross cladding loss and UO_2 contamination of the circuit.

It can be seen that commercial light water reactor fuel containing defects can be operated for hundreds of days before failure. Therefore spent fuel will contain very few failed rods, unless some exceptional operational conditions have occurred. The reviews by Locke has been followed by a review of recent German experience of operation with defective fuel rods (28). The German experience confirms the predictions by Locke.

It is well known that precipation of hydrides embrittles the Zircaloy cladding. But the failure mechanism at reactor operating temperatures is not one of brittle failure of the cladding. It is rather an extreme case of hydriding, where the whole cladding thickness has been transformed to hydride. Therefore it can be expected that among the category "successful defect operation" in Fig. 1 many, if not all of the defective rods, contain extensive amounts of hydrogen, which will make the rods brittle at low temperatures. A recent assessment of cladding ductility after irradiation (29) showed that for intact fuel rods where the hydrogen content was assumed to be at most 600 ppm, recrystallized cladding could be expected to survive a few tenths of a percent of plastic deformation before failure, whilst cold worked cladding fails without any plastic deformation. No information seems to exist for stress relieved cladding. Therefore a fuel bundle with intact fuel rods of recrystallized Zircaloy could be expected to survive a handling accident resulting in a permanent bow of the fuel rods (see section 3). Since there is no information

either on the hydrogen content and hydrogen distribution in defective fuel rods, or on the ductility of cladding at these hydrogen contents, there is a clear possibility that defective fuel rods are very brittle and easily damaged during handling and transport. Reports on increased damage during transport (4) seem to confirm this possibility.

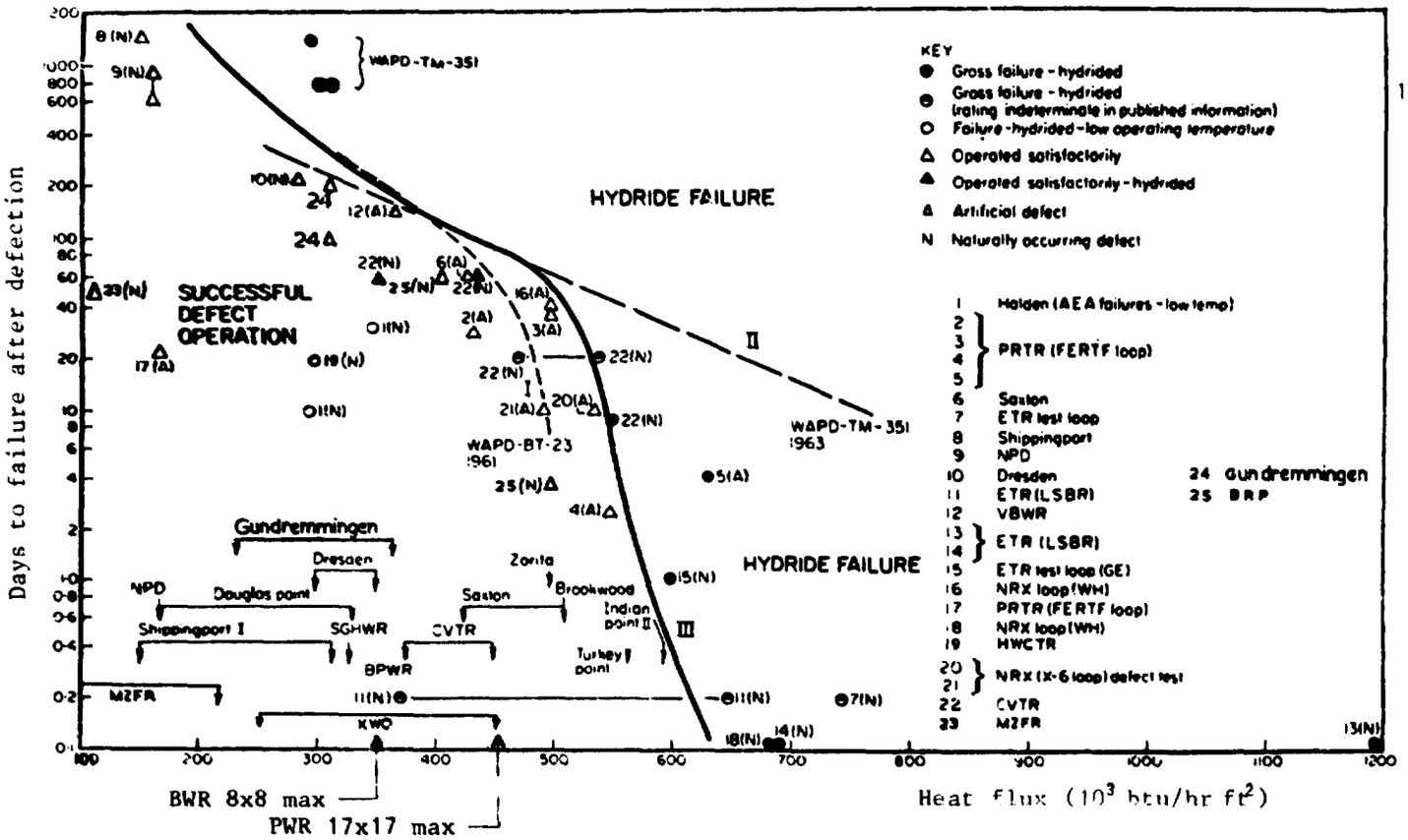

Fig. 1 Assessment of the failure of defective zirca'oy clad fuel from (26).

$1\ W/cm^2 = 3.17 \cdot 10^3\ btu/ft^2 \cdot h$

6 DEGRADATION MECHANISMS FOR DEFECTIVE FUEL RODS

Most potential degradation mechanisms for spent light water reactor fuel rods are common to intact and defective rods and therefore will not be considered here. They have been treated by both Johnson (1) and Vesterlund/Olsson (2). It can even be argued that a few potential degradation mechanisms, delayed hydride failure and fission product induced stress corrosion cracking, will be less severe in defective rods, since the driving force for these failure mechanisms, the internal pressure, is normally absent.

There are however a few special degradation mechanisms for defective rods. These are:

- internal hydriding
- internal oxidation accelerated by the presence of fission products
- oxidation of the UO_2, with or without fission product acceleration.

Internal hydriding must be regarded as relatively improbable. The basic mechanism should be similar to that described by Locke at reactor operating temperatures. Since the surface heat flux is extremely low for a fuel rod during storage it could in principle be possible to extrapolate Locke's curve to low heat fluxes. Unfortunately such an extrapolation would be extremely unreliable. But the basic processes behind internal hydriding involve drying the hydrogen gas diffusing up the fuel pin (26). This drying is caused by oxidation of cladding and UO_2.

Since these processes are extremely slow compared to steam diffusion in the fuel clad gap at storage temperatures, no drying will take place, and the moisture in the gas will heal any defects in the oxide and prevent the hydrogen from penetrating the cladding. The moisture will also

oxidize the surface of any sun-burst since it is known that zirconium hydride oxidizes more easily than Zircaloy (30). It is not known however whether or not this oxide film can serve as a barrier to hydrogen. It is therefore possible that growth of already formed sun-bursts may occur. The amount of growth will depend on the supply of hydrogen, which in turn is dependent on cladding and UO_2 oxidation and radiolysis of the water in the fuel rod.

Internal oxidation of the Zircaloy can only be a problem if fission products localize and accelerate the attack. There is convincing evidence that uniform corrosion in a water environment will not be a problem during pool storage (1, 2). Observations on failed fuel rods tested in the R2-reactor, indicate that oxide patches are sometimes formed on the inside of the fuel rods. Little is known about the rate at which these oxide patches form. It has been suggested that Cs is involved in the formation of these patches (33). Garzarolli et al (28) also observed formation of oxide patches on the inside of defective fuel rods, and explained them as a result of a local depassivation of the uniform oxide layer by fission products such as I and Cs. Since nothing is known about the temperature dependence of the growth rates of these patches, it must be assumed that they might grow during pool storage. Even if they do not threaten the integrity of the cladding, they provide a source of hydrogen which contributes to the growth of sun-bursts.

Oxidation of the UO_2 results in transformation of the UO_2 to U_3O_8 and a volume expansion of the fuel pellet. Such oxidation occurs during operation of defective fuel rods, but the reaction rate is slow. At small defects (pinholes), where there is no direct contact with the coolant flow, the U_3O_8 layer is thin (10-30 μ) even after one year of operation. At larger defects the layer may grow to about 1000 μ

after one year of exposure (28). Vesterlund/Olsson (2) extrapolated literature data on the oxidation of UO_2 to pool storage temperatures, and concluded that the growth of U_3O_8 would be about $5 \cdot 10^{-6}$ mm/100 years. It is clear that such an extrapolation is very uncertain since it does not include an effect of fission products. Vesterlund/Olsson also suggested that it was uncertain whether or not U_3O_8 could form at pool storage temperatures. From literature data they assumed that U_2O_7 would be the stable oxide at low temperatures. With the formation of U_2O_7 no volume expansion is expected, since U_2O_7 has nearly the same density as UO_2. However, there is another consequence of a phase transformation: the transformation may result in the release of the fission products trapped in the UO_2 matrix (6). It would therefore be interesting to study the oxidation of UO_2 regardless of which stable oxide forms at low temperatures.

At the Windscale Inquiry, Flowers presented results on an examination of a defective fuel rod which had been stored for 9 years (34). A diameter measurement showed that no measurable swelling of the Zircaloy cladding, caused by pellet expansion during pool storage, had occurred. It must be pointed out that this observation does not exclude the possibility that oxidation of the UO_2 has occurred. It is even possible that the cladding is stressed in the elastic range by the pellet, so that delayed hydride cracking or fission product induced stress cracking could occur.

In summary: special degradation processes for defective fuel exist; they may well be considered unlikely, but there is no proof that they cannot occur during pool storage for 10 - 60 years. There is therefore a need for work both on the examination of defective fuel rods and on basic mechanisms.

7 RECOMMENDATIONS

7.1 Recommendations based on current knowledge

All experience shows that there is no particular problem with pool storage of defective fuel, and that any long term degradation effects which may work seem improbable.

There is a possibility that defective fuel rods are hydrided and embrittled so that increased damage may occur during transport.

It is possible to identify and remove defective fuel rods at the reactor pools. Experience with repairing defective bundles exists and is more or less a routine operation.

With the above facts in mind it seems as if a sensible course of action would be to identify defective fuel rods and remove them from the bundles before the bundles are transported to the facility for interim storage. One way of dealing with the defective rods would be to place them in a suitably sealed Zircaloy tubes. To minimize the possibility of hydriding of the outer tube it should probably be preautoclaved, or have some other suitable surface treatment. Any corrosion resistant material could be used for the outer tube, but perhaps Zircaloy would have the least influence on future reprocessing of the defective fuel.

After recladding the defective fuel rods can be collected in bundles and transported to the storage facility. The risk of transport damage should be less than for intact rods.

7.2 Future research related to pool storage of defective fuel

There is a possibility that our recommendation is inadequate: if there is a large group of defective hydrided rods which give no failure indication in the sipping test. Then action according to our recommendation only

solves a part of the problem. On the other hand the problem with transport damage due to hydriding may have been overestimated. Therefore investigations on the hydriding and brittleness of defective fuel rods should be undertaken. If these investigations show that brittleness of the cladding of the defective fuel rods is indeed a problem, then there is some incentive for carrying out some eddy current tests on rods from bundles without any sipping indication, in order to get an idea about the magnitude of the problem with non-leaking defective rods.

With regard to potential degradation mechanisms for defective rods storage experience shows that nothing drastic happens. Therefore there is time to perform studies on the degradation mechanisms, so that basic data relevant to pool storage are produced. It can be argued of course, that recladding of individual fuel rods makes studies of degradation mechanisms unnecessary, but under special circumstances (such as fuel failure rates much higher than the present values of 10^{-4} to 10^{-3} (33)) recladding of individual fuel rods may be impractical.

Studies of degradation mechanisms should also include the release of fission products due to UO_2-oxidation.

REFERENCES

1. A B Johnson, Jr
Behaviour of spent nuclear fuel in water pool storage.
BNWL-2256, Battelle Pacific Northwest Laboratories, 1977.

2. G Vesterlund and T Olsson
Degraderingsmekansimer vid bassänglagring och hantering av utbränt kraftreaktorbränsle.
RB 78-29, ASEA-ATOM 1978.

3. A B Johnson, Jr
Impacts of reactor-induced cladding defects on spent fuel storage.
PNL-SA-6917. Battelle Pacific Northwest Laboratories 1978, Presented at the NEA Seminar on the Storage of Spent Fuel Elements, Madrid, June 20-23, 1978.

4. S Hagsgård
Private communication

5. E A Grimm
Design and operational characteristics of high density fuel storage facilities.
Proc. from the NRC-IAEA spent fuel storage meeting, Febr. 28 - March 2, 1978 (NUREG-0488)

6. B F Warner
The storage in water of irradiated oxide fuel elements.
Presented to the Windscale Public Inquiry 1977.

7. K L Huppert and G Zimmermann
Erfahrungen mit der Langzeitlagerung von LWR-brennelementen, Reaktortagung 1977 p. 447

8. M Peehs et al
Zwischenlagerung abgebrannter LWR-brenn-elemente: Erfahrungen und Aufgaben aus der Sicht der Brennelemente
Reaktortagung 1978, p. 421

9. R Forsyth
Post failure fuel performance program (PFFP)
Studsvik Arbetsrapport K4-78/6, Studsvik Energiteknik AB, 1978.

10. E C Tarnuzzer
Spent fuel shipping experience - Yankee Atomic Electric Company Trans. Am. Nucl. Soc. 16 (1973), p. 174.

11. H Keese
Experience in radioactive material shipments in Europe.
Proc. Int. Symp. on the Management of wastes from the LWR fuel cycle, Denver, Colorado July 11-16, 1976, p. 429.

12. J W Langhaar
Transport experience with radioactive materials
Ibid. p. 469.

13. P Rübenberg, R Brunner, K Knecht, G Kaspar
Lokalisierung von Brennelementschäden beim Erstkern von KKS
Reaktortagung, 1974, p. 324.

14. H Knaab, K Knecht, R Manzel, E Schuster, H Stehle
Untersuchung des Betriebsverhaltens von Brennelementen des Kernkraftwerkes Obrigheim (KWO)
Reaktortagung, 1972, p. 221.

15. F Garzarolli, H Knaab, K Knecht
KWU Brennelement-Service. Stand der Entwicklung und neuere Ergebnisse
Reaktortagung, 1976, p. 462.

16. R Manzel, H Stehle
KWU in-reactor experience with LWR fuel
Nuc. Eng. Mat. Vol. 3
Proceedings Paris, 21-25 April, 1975.

17. M. de Donder, J Privot
Detection of failed fuel subassemblies by the sipping technique
Reaktortagung, 1971, p. 453.

18. T A Green
Detection of detective fuel using vacuum sipping technique
Trans. Am. Nucl. Soc. 30 (1977) p. 333-5

19. M Skafi
Erfahrungen mit dem Sipping Verfahren und Aussagen über dessen Zuverlässigkeit für die Schadenfeststellung an BE von SWR:s
Reaktortagung, 1974, p. 328.

20. G Rönnberg
Private communication

21. U Bergenlid
Private communication

22. H Knaab, K Knecht and F Garzarolli
Kraftwerk Union (KWU) nuclear fuel service
Kerntechnik 19 (1977) p. 122.

23. C Oreland
Private communication

24. K Pettersson, G Vesterlund and T Andersson
The effect of irradiation on the strength, ductility and defect sensitivity of fully recrystallized Zircaloy tube.
Studsvik 78/5, Studsvik Energiteknik AB, 1978.

25. D H Locke
Defected Zircaloy fuel
Nuclear Engineering International, August 1969, p. 648.

26. D H Locke
The behaviour of defective reactor fuel
Nuclear Eng. Design 21 (1972) p. 318.

27. D H Locke
Defective fuel behaviour in water reactors
ANS Topical Meeting on Water Reactor Fuel Performance, May 9-11, 1977
St. Charles Ill., USA.

28. F Garzarolli, R Manzel and H Stehle
The behaviour of defective fuel rods under continued reactor operation.
Kerntechnik 20 (1978), p. 463.

29. K Pettersson
Mekaniska kapslingsskador vid långtidsförvaring och hantering av utbrända bränslestavar.
TPM-MF-568, AB Atomenergi 1975.

30. R C Asher and F W Trowse
The distribution of hydrogen in zirconium-alloy fuel cladding: The effects of heat flux.
J. Nuclear Materials 35 (1970), p. 115

31. G Lysell
Private communication

32. R H Flowers
Results of an examination of irradiated oxide fuel following storage in water.
Evidence given at the Windscale Inquiry.

33. D H Locke
Review of experience with water reactor fuels 1968 - 1973
Nuc. Engin. Design, Sept. -75, Vol. 33(2).

STUDSVIK/K4-79/28

CRITERIA FOR RECLADDING OF SPENT LIGHT WATER REACTOR FUEL BEFORE LONG TERM POOL STORAGE

SKI B56/78, Final report

Kjell Pettersson and Lennart Jansson

Studsvik

Studsvik Energiteknik AB S-611 82 NYKÖPING
SWEDEN Telefon 0155-200 00 Telex 64013 studs s