July 8, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE ENERGY CORPORATION	)	Docket Nos. 50-413-OLA
	)	50-414-OLA
	)	
(Catawba Nuclear Station	)	
Units 1 and 2)	)	

NRC STAFF REBUTTAL TESTIMONY OF DR. RALPH LANDRY AND DR. RALPH O. MEYER CONCERNING BREDL CONTENTION I

Q1. Please state your name.

A1a. (RL) My name is Ralph Landry. I am a Senior Reactor Engineer employed by the NRC in the Office of Nuclear Reactor Regulation. A statement of my professional qualifications was attached to the NRC Staff Testimony of Undine Shoop, Dr. Ralph Landry and Dr. Ralph O. Meyer Concerning Contention I, (Staff Prefiled Testimony),[1] filed July 1, 2004.

A1b. (ROM) My name is Ralph O. Meyer. I am employed as a Senior Technical Advisor for Core Performance and Fuel Behavior in the Office of Nuclear Regulatory Research at the NRC. A statement of my professional qualifications was attached to the Staff Prefiled Testimony, filed July 1, 2004.

[1] The NRC Staff submitted several proposed exhibits with the Staff Prefiled Testimony. The Staff intends to introduce only the following portions of the documents into evidence: Exhibit 3, "LOCA Test Results for High-Burnup BWR Fuel and Cladding," Y. Yan, et al., pages 1, 17 (unpaginated); Exhibit 4, "Does M5 Balloon More Than Zircaloy-4 Under LOCA Conditions?," N. Waeckel, et al., pages 1-2, 10; Exhibit 5, Memorandum from F. Eltawila to S. Black, "Response to User Need for Development of Radiological Source Terms for Review of Mixed Oxide Fuel Lead Test Assemblies," February 23, 2004, pages 1-2 and Attachment B, page B-5, Figure 1.

Q2. What is the purpose of this testimony?

A2. The purpose of this rebuttal testimony is to address the Prefiled Written Testimony of Dr. Edwin Lyman Regarding Contention I, submitted on behalf of the Blue Ridge Environmental Defense League (BREDL) on July 1, 2004.

Q3 In the answer to Question 5 of his prefiled written testimony, Dr. Lyman states that "Appendix K to Part 50...sets forth ECCS 'evaluation models', *i.e.* assumptions about the behavior of reactor fuel that are to be used in determining whether it complies with the criteria in 10 C.F.R. § 50.46." Is this statement correct?

A3. (RL) No, this is not a correct statement. Appendix K to 10 C.F.R. Part 50 provides the descriptions of the required and acceptable features of the evaluation models as well as the required documentation. Appendix K does not provide assumptions about the behavior of reactor fuel. The topics covered include the sources of heat during the LOCA, the swelling and rupture of the cladding and fuel rod thermal parameters, the blowdown phenomena, and the post-blowdown phenomena. These are all descriptions and specifications placed on phenomena that must be modeled by an acceptable evaluation model. The majority of the specifications provided by Appendix K deal with thermal hydraulic phenomena, that is, heat transfer and fluid flow behavior.

More specifically, only the first few paragraphs of Appendix K address matters related to fuel, such as how the decay heat is to be calculated, how stored energy is to be calculated, and how the heat from the reaction of the cladding material with the cooling water, or steam, is to be calculated. The remainder of the appendix gives specific details and requirements on how the heat removal by the coolant water is to be calculated, and how the movement of the coolant water through the reactor system is to be calculated. Thus, the majority of Appendix K provides details on how cooling of the reactor fuel is to be calculated, along with details on how the movement of the cooling water around the system is to be calculated. Only the early part of the appendix is concerned with specifying how the amount of heat contained and produced by the fuel are to be calculated.

Q4 In his answer to Question 18 of his prefiled written testimony, Dr. Lyman states that "...the Staff claims to have independently verified the adequacy of Duke's LOCA analysis..." Is this statement correct?

A4. (RL) No, this is not a correct statement. The staff does not claim to have performed independent analyses or calculations to verify the submittal of Duke. The staff has stated in its Safety Evaluation (SE): "Based on the NRC review of the information provided, the NRC staff concludes that the effect of four MOX LTAs has been conservatively evaluated and has been demonstrated to be in compliance with the requirements of 10 C.F.R. 50.46." That statement was based on the material submitted by Duke that was reviewed and found to be consistent with the approval the staff has granted with regard to the fuel vendor's LOCA evaluation model and effect on the LOCA analysis of record for the Catawba Nuclear Plant.

Q5. In the answer to Question 6 of his prefiled written testimony, Dr. Lyman explains why he thinks the Appendix-K evaluation models should include consideration of fuel relocation during LOCAs. He bases his explanation partly on your memorandum of February 8, 2001, and on Mr. Thadani's memorandum of June 20, 2002. Do you agree with his response?

A5. (ROM) No. Appendix K has been in effect since 1974, and over the years some extra conservatisms and some non-conservatisms have been identified. To the best of my knowledge, the NRC never contemplated including fuel relocation in Appendix K as mentioned by Dr. Lyman. However, by using Appendix K with its compensating extra conservatisms, Duke has adequately accounted for any non-conservatisms. Furthermore, based on my experience and knowledge, and as is demonstrated in my testimony, I do not agree that certain characteristics of MOX fuel exacerbate the effects of fuel relocation.

Q6. In the answer to Question 12 of his prefiled written testimony, Dr. Lyman states that "Tight bonding has also been observed at the Halden reactor in Norway to retard the rate of balloon formation." Is that statement correct?

A6. (ROM) No. Dr. Lyman accurately interprets a statement made in NEA/CSNI/R(2003)9, but the statement in that report is not correct. No ballooning tests with high burnup fuel rods have been performed at Halden as of this date, and the statement in NEA/CSNI/R(2003)9 was merely a suggestion of what might happen rather than a report of what has been experimentally observed. I have verified that this statement is in error by an e-mail exchange with Dr. Wolfgang Wiesenack, who is the general manager of the Halden Project (Exhibit A, Wolfgang Wiesenack e-mail to Miroslav Hrehor, "Re: Statement in one of our SEGFSM Reports," 6/14/04). Miroslav Hrehor, who is also mentioned in that e-mail, is the scientific secretary at NEA who is responsible for that report.

Q7. In the answer to Question 12 of Dr. Lyman's prefiled written testimony, he states that "It has been confirmed that MOX fuel is more resistant to clad failures due to pellet-clad mechanical interaction (PCMI) than LEU fuel, even at high burnups." Is that observation relevant to the behavior of MOX fuel under LOCA conditions?

A7. (ROM) No. First, there is no PCMI during a LOCA. PCMI occurs when the power is increased and thermal expansion of the pellet, which is greater than that of the cladding, causes the pellet to push on the cladding. During a LOCA, power is decreased and the cladding expands faster than the pellet -- actually moving away from the pellet. More fundamentally, though, the additional resistance of MOX fuel to cladding failure by PCMI is the result of the greater plasticity of the MOX pellets. They are softer than LEU pellets. The MOX fuel pellets are thus able to deform somewhat and relax the stress they apply to the cladding. This has nothing to do with bonding between the pellets and the cladding.

Q8. In the answer to Question 12 of Dr. Lyman's prefiled written testimony, he claims that LOCA test results from the Power Burst Facility (PBF) have shown that irradiated rods experience greater deformation (swelling) than unirradiated rods. He then states that there is no way to determine whether Duke's LOCA analysis underestimates or overestimates the degree of cladding swelling. Do you agree with this conclusion?

A8. (ROM) No. As discussed by the IRSN authors in the reference cited by Dr. Lyman (Mailliat and Schwartz at 432), it would appear that the PBF tests showed an enhancement of more than a factor of 2 in balloon size for irradiated rods compared with unirradiated rods. This was said to be the result of more uniform temperatures in the irradiated rods. This is probably not an accurate interpretation of the test results. After the PBF tests were performed, more work was done on the effects of temperature uniformity at the Karlsruhe nuclear research center in Germany by F.J. Erbacher and coworkers. In a review article by Erbacher, the PBF results are discussed along with additional German test results with irradiated and unirradiated rods (Exhibit B, F. J. Erbacher, "Cladding Tube Deformation and Core Emergency Cooling in Loss of Coolant Accident of a Pressurized Water Reactor," *Nuclear Engineering and Design*, 1987, pp. 55-64). In Figure 2 of this paper, the PBF data are plotted along with other data, and the temperature at the time of bursting for the PBF data is seen to vary from about 800 to 1100 C. Also shown in this figure are curves that show the strong variation of balloon size associated with the temperature at the time of bursting. This well known variation in balloon size is the result of changes in the crystal structure of the cladding, which switches from an alpha-phase to a beta-phase between 800 and 1000 C. The rod that produced the largest balloon ruptured in the alpha phase just below 800 C, right at a temperature where balloon sizes are expected to be their maximum. Thus, it was probably the differences in temperature at the time of rupture of the PBF rods that produced the different balloon sizes rather than the difference in burnup. In Figure 5 of Erbacher's paper, results are shown for a substantial number of ballooning test in the same facility. No systematic difference is seen between fresh fuel and the irradiated rods. High-burnup effects are being studied in the ongoing NRC research program to further clarify LOCA behavior, but it can be noted again that the variation in balloon sizes just discussed is not related to the use of MOX fuel pellets.

Q9. In the answer to Question 14 of Dr. Lyman's prefiled written testimony, he concludes that the recent Electricité de France (EDF) presentation at Argonne National Laboratory does not fully address the differences in the size of balloons between M5 and Zircaloy cladding. Do you agree with that statement?

A9. (ROM) I agree that the EDF presentation does not entirely address the differences in the size of balloons between M5 and Zircaloy, but it clearly shows that the large difference claimed by IRSN is a consequence of using inappropriate data. Further, Dr. Lyman's comment about spalling, or flaking of a thick oxide coating, is not relevant. To the best of my knowledge, none of the ballooning tests utilized cladding with spalled oxide, and certainly no spalling is expected in the Catawba core with its modern cladding materials (ZIRLO and M5). With regard to the size of the balloons, it should not be forgotten that ballooning is an M5 cladding issue; it is not a MOX issue. Based on my knowledge and experience, and the testimony I have given, there is no valid reason to expect that the size of the balloons will be affected by the type of fuel inside. Although confirmatory research on M5 cladding under LOCA conditions is continuing, it is my opinion that the specific concerns raised by Dr. Lyman are not valid. The staff believes the ballooning size has been adequately accounted for in the analysis.

Q10. In the answer to Question 15 of Dr. Lyman's prefiled written testimony, he quotes an IRSN presentation as saying, "The impact of fuel relocation in fuel rod balloons ... is still fully questionable and should be addressed by specific analytical tests with a simulation of fuel relocation." Do you agree?

A10. (ROM) No. As discussed in my answer to Questions 33 and 34 of the Staff's Prefiled Testimony, the diameter of the balloons will not be different for MOX fuel and LEU fuel. Further, the diameter of the balloons will not be affected by fuel relocation because fuel relocation would occur after the balloons are formed. Thus neither MOX fuel nor fuel relocation will affect flow blockage, which is calculated by the models used by Duke. Therefore, the Duke analysis is not incomplete and is not likely to be non-conservative.

Q11. In the answer to Question 16 of Dr. Lyman's prefiled written testimony, he points out that Duke's calculations have demonstrated a peak cladding temperature (PCT) that is higher for a MOX fuel rod than for an equivalent LEU rod in the same position in the core. He then concludes that the margin is therefore smaller for a MOX rod than for an LEU rod in the same position. Do you agree with that conclusion?

A11. (ROM) No. Duke's analysis shows a higher PCT for MOX fuel only because they used the same decay heat curve for MOX and for LEU fuel. In fact, the decay heat is lower for MOX fuel than for LEU fuel at the time of importance for LOCA (Exhibit C, "Decay Heat Power in Light Water Reactors," ANS standard ANS/ANS-5.1-1994, pp.1, 2, 14, 16). Therefore, in reality, the PCT for MOX fuel should be a little lower than the PCT for LEU fuel and the margin will not be reduced.

Q12. In the answer to Question 17 of Dr. Lyman's prefiled written testimony, he states that the only way to fully address the uncertainties associated with the behavior of high-burnup, M5-clad MOX fuel during LOCAs is to conduct integral LOCA tests of such fuel. Do you agree with that statement?

A12. (ROM) No, I do not agree that integral LOCA tests of high-burnup MOX fuel with M5 cladding are needed. The effect of plutonium on LOCA behavior is almost entirely the result of small changes in initial stored energy, fission heat, and decay heat as discussed in the Staff's prefiled testimony in answer to Questions 17, 18, and 24. These changes are well known and adequately modeled. There has been speculation that MOX fuel would enhance the effect of fuel relocation into balloons during a LOCA, but it is my opinion that there will be no such enhancement (see answer to Question 40 in the Staff's prefiled testimony). It has also been claimed that cladding behavior will be altered by MOX fuel in comparison with LEU fuel, but I have offered testimony that shows there will be no effect of the type of fuel pellets inside on cladding behavior (see answer to Question 34 in the Staff's prefiled testimony). In my opinion, the uncertainties associated with the MOX LTAs for Catawba are adequately understood.

Q. Does this conclude your REBUTTAL testimony?

A. Yes.

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(Catawba Nuclear Station	)	
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CERTIFICATE OF SERVICE

I hereby certify that copies of the "NRC STAFF REBUTTAL TESTIMONY OF DR. RALPH LANDRY AND DR. RALPH O. MEYER CONCERNING BREDL CONTENTION 1" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class; or as indicated by an asterisk(*), by deposit in the Nuclear Regulatory Commission's internal mail system; and by e-mail as indicated by a double asterisk(**), this 8th day of July, 2004.

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EXHIBIT A

From: Wolfgang Wiesenack <wowi@mail.hrp.no>
To: <Miroslav.HREHOR@oecd.org>, <ROM@nrc.gov>
Date: Mon, Jun 14, 2004 8:03AM
Subject: Re: Statement in one of our SEGFSM Reports

Ralph,

I agree that your formulation is more precise and unambiguous. The original addition "thus restraining the rate of clad ballooning" may be seen as stating something experimentally verified or as an inference (the latter was intended). But I also wonder what the difference would be between a 400cm and a 40cm rod.

Miroslav: I suggest to change the document accordingly.

Wolfgang

===

> Date: Wed, 09 Jun 2004 16:09:55 -0400
> From: "Ralph Meyer" <ROM@nrc.gov>
> To: <wolfgang.wiesenack@hrp.no>, <Miroslav.HREHOR@oecd.org>
> Cc: "Harold Scott" <HHS.twf5_po.TWFN_DO@nrc.gov>
> Subject: Statement in one of our SEGFSM Reports
>
> Wolfgang, Miroslav,
>
> On p. 79 of "Ongoing and Planned Fuel Safety Research in NEA Member States" it says:
>
> "Halden have carried out axial gas flow studies in fuel rods over a range of burn-up and test have shown a severe restriction in volume flow at high burn-up thus restraining the rate of clad ballooning."
>
> I think the last part of this statement is not correct. I know that Halden has carried out axial gas flow studies that show severe restriction in flow at high burnup, but I don't think any of these were tests with LOCA clad ballooning. Shouldn't the statement say "... thus suggesting that the rate of clad ballooning might be restrained."?
>
> This statement has been quoted by Lyman in the MOX hearing that is underway here.
>
> Ralph
>
> P.S. I will be away from the office until Tuesday, June 15
>
>

CC: <wolfgang.wiesenack@hrp.no>

EXHIBIT B

Nuclear Engineering and Design 103 (1987) 55–64
North-Holland, Amsterdam

CLADDING TUBE DEFORMATION AND CORE EMERGENCY COOLING IN A LOSS OF COOLANT ACCIDENT OF A PRESSURIZED WATER REACTOR

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Received 4 November 1986

The paper summarizes the dominant effects which finally ensure the core coolability of a pressurized water reactor in a loss-of-coolant accident (LOCA).
The main results are summarized as follows:
- The cooling effect of the two-phase mixture which is intensified during reflooding increases temperature differences on the cladding tube circumference and thus limits the mean circumferential burst strains to values of about 50%.
- An unidirected flow through the fuel rod bundle during the refill and reflooding phases causes maximum cooling channel blockage of about 70%.
- The coolability of deformed fuel elements can be maintained up to flow blockages of about 90%.

All effects investigated indicate that in a LOCA no impairment of core coolability and public safety has to be expected.

1. Introduction

In the licencing procedure under the Atomic Energy Act evidence must be provided that the consequences of all conceivable pipe ruptures in the primary circuit resulting in loss of coolant can be controlled. For these so-called loss-of-coolant accidents the double rupture of a main coolant line between the main coolant pump and the reactor pressure vessel is presently considered as the design basis for core emergency cooling systems.

After rupture of a main coolant line the reactor is shutdown automatically, even without actuation of the shutdown rods. But the decay heat still generated after suspension of the chain reaction necessitates reliable long-term cooling of the fuel element cladding tubes. This is achieved by core emergency cooling systems which, after evacuation of the reactor pressure vessel, feed into the reactor core the borated emergency cooling water stored in accumulators and pools so as to cover the reactor core again with coolant and ensure reliable long-term cooling of the fuel elements.

However, before emergency cooling becomes fully effective, fuel element cooling deteriorates temporarily. Zircaloy fuel rod claddings may attain temperatures at which they balloon or burst under the impact of the internal overpressure. This narrows locally the coolant channels. Further damage to the fuel elements can be prevented only if the emergency cooling systems, despite the reductions in flow sections, guarantee reliable cooling of the fuel elements and no further major rise in temperature occurs.

Within the framework of safety analysis and licensing procedures, providing evidence for the following items is of particular importance: number of burst fuel rod cladding tubes, size of the burst circumferential strain, axial displacement of the burst points, maximum coolant channel blockage and coolability of deformed fuel elements.

The research activities conducted under the Nuclear Safety Project (PNS) by various institutes of the Karlsruhe Nuclear Research Center (KfK) served the primary purpose of elaborating the relevant experimental and theoretical fundamentals. The out-of-pile and in-pile experiments started from the design data of the emergency cooling systems and the fuel elements for pressurized water reactors built by Kraftwerk Union (KWU). The Zircaloy-4 cladding tubes used were in conformity with the KWU specification; they had been cold worked and stress relieved; the external diameter was 10.75 mm, the inner diameter 9.3 mm.

It has been proved in the COSIMA experiments that under realistic boundary conditions of a loss-of-coolant accident no noticeable cladding tube deformations have to be expected in the blowdown phase [1]. Therefore, only the refill and reflooding phases are important for deformation and coolability.

In the following sections some of the most important results will be compiled of fuel rod and fuel element behavior during the refill and reflooding phases of a loss-of-coolant accident.

2. Deformation mechanism of Zircaloy cladding tubes

In order to record the consequences both of a rupture of the main coolant line (design basis accident) and of incidents involving small leaks (small load due to differential pressure) and to take account of developments resulting in an increase in target burnup (high load due to differential pressure), the deformation and burst behavior was investigated for a wide pressure range in single-rod experiments. However, the investigations concentrated on the range of burst pressure of about 50 to 70 bar which must be supposed in a design basis accident of pressurized water reactors.

The following figures show the experimental results compared with the calculated values. The calculated values traced as curves has been obtained with a computer code developed under the REBEKA task [2].

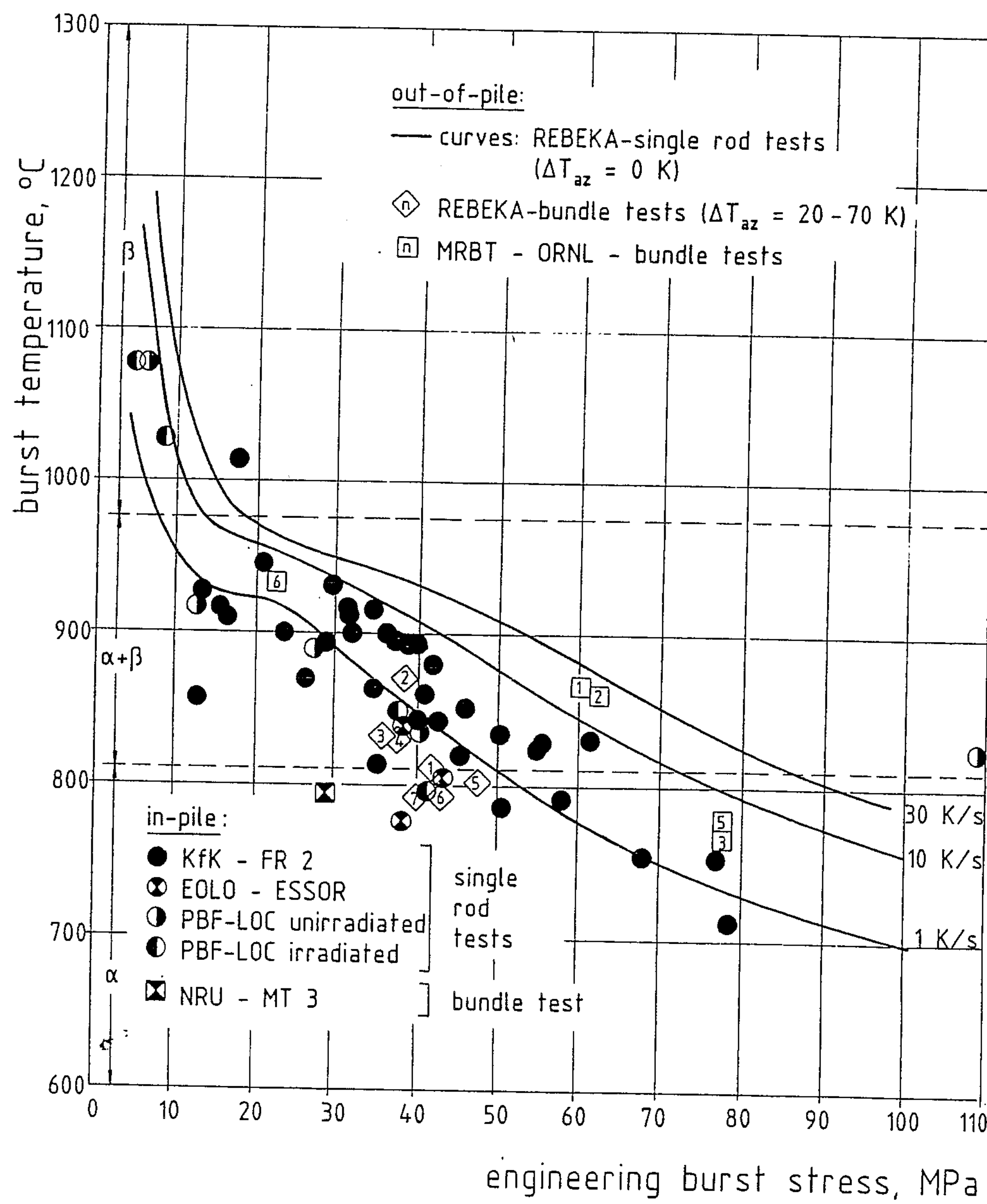


Fig. 1. Burst temperature vs. burst stress of Zircaloy claddings.

2.1. Burst temperature

Experimental data for the burst temperature among other factors provide an important basis for the number to be determined of burst cladding tubes and for the fission product release resulting from them.

Fig. 1 shows the burst temperature versus the engineering burst stress with the heating rate as the parameter. At the same heating rate a higher rod internal pressure causes the burst temperature to become lower. The results of the REBEKA single rod tests represented as plots show a marked influence of the heating rate on the burst temperature, i.e., high heating rates lead to higher burst temperatures than low heating rates. Similar experiments performed in the FABIOLA testing facility have confirmed the relationships described and, moreover, have shown that fission products simulated by iodine do not exert an influence on the deformation behavior [3].

The same figure shows a comparison with out-of-pile bundle tests and with various in-pile tests. Taking into account the differences in the experimental conditions and the difficulty of determining the burst temperature exactly, the agreement of all experimental data can be termed good. No influence of nuclear parameters on the burst temperature has been found [4,5].

Therefore, it can be assumed that with this information the number of defective fuel rods can be determined with adequate accuracy in a loss-of-coolant accident if the temperature and pressure development of the fuel rods is known. Accordingly, with the present inner pressures and burnups, the cladding tubes will fail through burst when temperatures of about 800° C are attained.

2.2. Burst circumferential strain

The scope of burst circumferential strain of the Zircaloy cladding tubes determines inter alia decisively the coolant channel blockage and the coolability in the fuel element.

Fig. 2 shows the burst circumferential strain versus the burst temperature with the heating rate as the parameter. The calculated values traced as curve describe the burst circumferential strains measured in

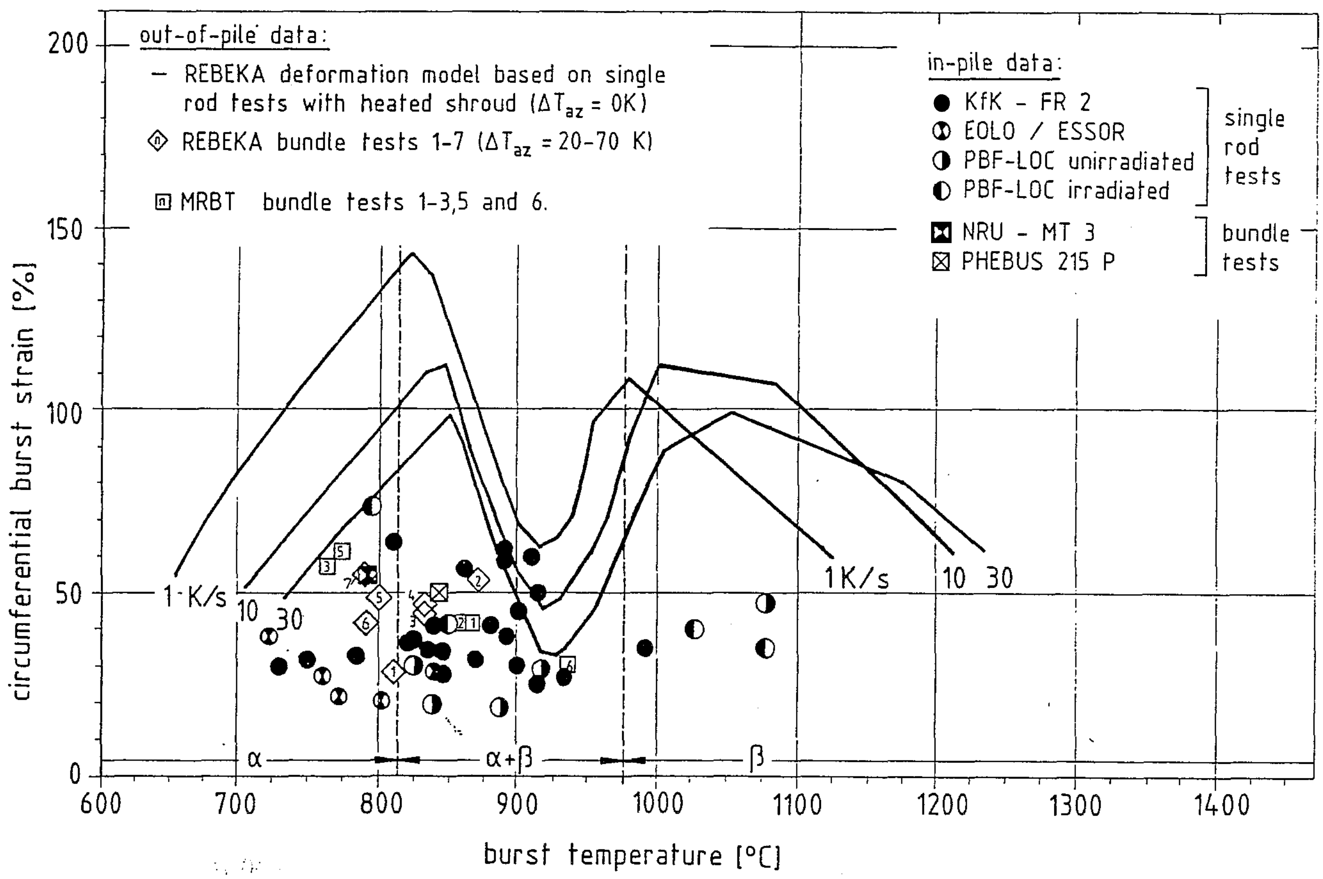


Fig. 2. Burst strain vs. burst temperature of Zircaloy claddings.

REBEKA single rod tests on the cladding tube circumference at uniform temperature. The strain maxima at 820 °C and approx. 1000 °C can be attributed to the superplasticity of Zircaloy [6]. They occur to a remarkable extent only at nearly uniform temperature on the cladding tube circumference and with symmetric deformation of the cladding tube. These idealized conditions were specifically provided by a heated tube in the neighborhood in order to have a systematic and fundamental experimental study performed of the deformation behavior in single rod experiments.

The averaged values from out-of-pile bundle tests and in-pile single rod and bundle tests entered in the figure indicate a marked reduction in the burst circumferential strains to values around 50%. This limitation is due to temperature differences on the cladding tube circumference. Lowering of the burst circumferential strain due to failure by embrittlement as a result of stress corrosion cracking has not to be expected under the boundary conditions of a loss-of-coolant accident [7]. The measured values of out-of-pile and in-pile tests entered in the figure do not suggest any impact of nuclear boundary conditions on the burst circumferential strain.

Under representative thermohydraulic boundary conditions of a loss-of-coolant accident heat flows from the pellet through the gap to the cladding tube and coolant are clearly established in a fuel rod. Tolerances in the dimensions of the pellets and cladding tubes as well as eccentricities of the pellets in the cladding tube lead to differences in gap widths along the cladding tube circumference and, consequently, to different heat transfer coefficients in the gap between the pellets and the cladding tube. In case of external cooling this causes temperature differences on the cladding tube circumference (azimuthal temperature differences).

In REBEKA single rod tests in which temperature differences were allowed to develop on the cladding tube circumference it has been proved that in case of deformation of Zircaloy cladding in the α- and early $(\alpha+\beta)$ phases of the Zircaloy a systematic relationship exists between the burst circumferential strain and the azimuthal temperature difference: Small azimuthal temperature differences cause a relatively uniform reduction in cladding tube wall thickness on the circumference and give rise to relatively high burst circumferential strains; great azimuthal temperature differences result in a preferred reduction of wall thickness on the hot part of the cladding tube circumference and to relatively low burst circumferential strains.

Fig. 3 shows in quantitative terms the dominant influence of azimuthal temperature differences on the burst circumferential strain. These relationships can be explained by bowing of the Zircaloy cladding tubes observed in a number of experiments in the α- and in the $(\alpha+\beta)$ ranges in case of deformation and azimuthal temperature differences. Tube bowing represented in fig. 4 produces the effect that the gap between the pellet and the cladding tube closes on the hot side and opens on the opposite cold side. This causes the azimuthal temperature differences to become larger during cladding tube deformation.

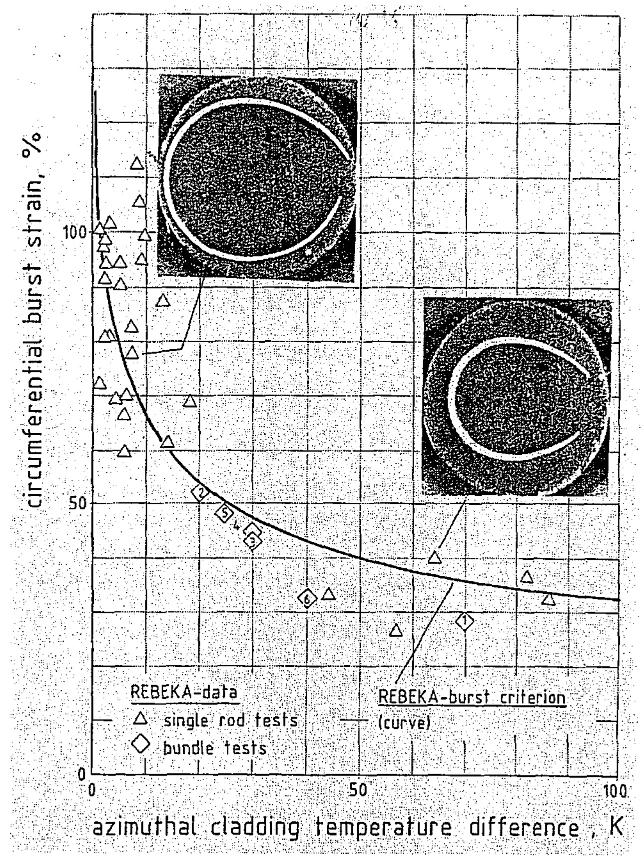


Fig. 3. Burst strain vs. azimuthal temperature difference of Zircaloy claddings.

This deformation behavior of Zircaloy cladding tubes is caused by the texture produced in cladding tube fabrication in the hexagonal, densely packed structure. The majority of hexagonal prism shaped crystals have their longitudinal axis and their prism planes oriented parallel to the cross sectional plane of the cladding tube. During plastic deformation under internal overpressure the strain behavior of these structures is anisotropic; this is characterized by the fact that the tube resists weakening of the wall thickness and, consequently, axial material flow takes place into the deformed zone which is paralleled by shortening of the

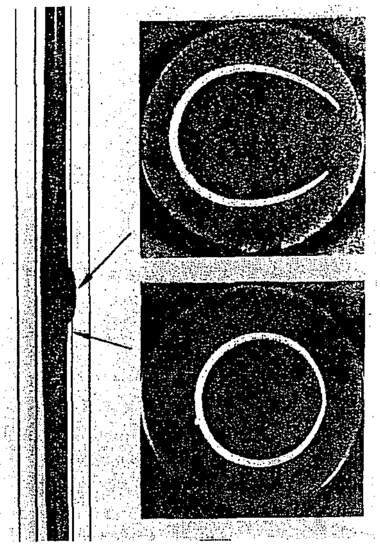


Fig. 4. Bowing of Zircaloy tubes during deformation under azimuthal temperature differences and cooling.

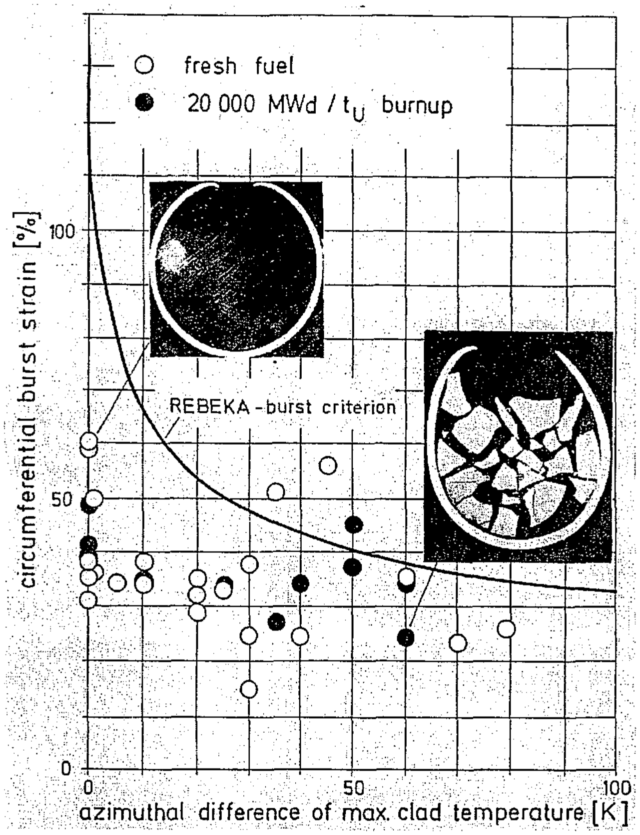


Fig. 5. FR 2 in-pile tests: Burst strain vs. azimuthal temperature difference of Zircaloy claddings.

tube in the course of deformation. If the wall thickness is weakened due to azimuthal temperature differences preferably on the hot side of the cladding tube, axial material flow and tube shortening are intensified on this same side. This causes bowing of the tube which implies that the hot cladding tube side contacts the pellets and the opposite colder cladding tube side moves away from the pellets. This is the reason why the deformation continues on the hot side as weakening of the wall thickness. As only the hot part of the cladding tube circumference undergoes deformation, this results in relatively low circumferential strains of the burst Zircaloy cladding.

In representative deformation experiments temperature differences of 30 K on the average were measured on the cladding tube circumference at the time of burst. This reduces the burst circumferential strains to values less than 50%.

Anisotropic strain behavior of Zircaloy and reduction of burst circumferential strains by temperature differences on the cladding tube circumference were also observed in the FR 2 in-pile experiments. It is visible from fig. 5 that also in the course of nuclear fuel rod deformation substantial azimuthal temperature differences occur. No influence has been proved to exist of the fragmented fuel of burnt up rods and neither an influence of the degree of burnup on the deformation behavior of Zircaloy cladding tubes.

2.3. Sensitivity to temperature of the Zircaloy cladding tube deformation

In all experiments performed it has been found that the deformation and burst behavior of Zircaloy cladding tubes responds very sensitively to the cladding tube temperature and that even temperature differences of less than 10 K exert a decisive influence on the deformation behavior.

Fig. 6 shows calculated circumferential strains as a function of the time for constant cladding tube temperatures of 790°C, 800°C and 810°C at a constant tube inner pressure of 60 bar. The figure illustrates the extreme sensitivity to temperature of Zircaloy deformation. Differences of not more than 10 K in the cladding tube temperatures imply changes of the burst time by about 30 s.

Because of the efficiency of emergency cooling the time at maximum cladding tube temperatures is limited: even small temperature differences on the Zircaloy cladding tubes decide upon whether the tubes will burst after large ultimate strains or whether deformation at a

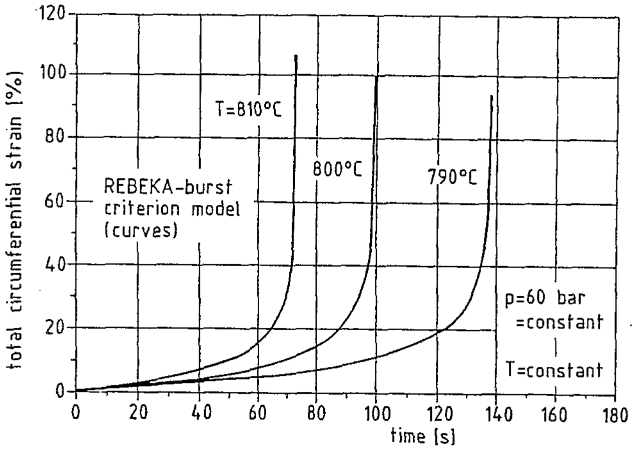


Fig. 6. Sensitivity of temperature of Zircaloy cladding tube deformation.

temperature plateau lower by about 10 K will cause the tubes to deform by just a few percent without burst. Even if the burst conditions are attained for all cladding tubes in a rod bundle, with the given unavoidable temperature differences the great differences in burst time prevent strong mechanical interactions from occurring between neighboring cladding tube and hence also greater deformation and damage propagation.

The high sensitivity to temperature of Zircaloy deformation makes evident that a precise deterministic prediction of cladding tube deformation in a fuel element is not possible on the basis of the thermohydraulics computer codes available today because the accuracy necessary for predicting cladding tube deformation of about 10 K cannot be achieved by these codes.

This underlines the importance of bundle tests to be performed under representative geometric and thermohydraulic boundary conditions so that the empirical information about the behavior of fuel elements in a loss-of-coolant accident which is needed for a scope of damage analysis can be derived.

3. Influence of thermohydraulics on cladding tube deformation and cooling channel blockage

In order to be able to assess the coolability of fuel elements in a loss-of-coolant accident burst experiments were performed on rod bundles in many countries. Very different burst strains and coolant channel blockages were found. These differences were considered for a long time as discrepancies not amenable to an explanation. However, it was supposed that they can be attributed to differences in the thermohydraulic boundary conditions of the experiments. Therefore, it had been the primary goal of the REBEKA bundle tests performed to study systematically the influence of thermohydraulics on cladding tube deformation.

It has been a general and important finding of the bundle tests that the deformation behavior of the Zircaloy cladding tubes in the rod bundle assembly follows the same laws of Zircaloy deformation as observed in the single rod experiments. The burst temperatures and burst pressures as well as the dependence of circumferential strain on the azimuthal difference of cladding tube temperatures agree well with the respective values from single rod tests (see figs. 1 and 3).

3.1. Influence of heat transfer on cladding tube deformation

It has been proved that the burst circumferential strain of the Zircaloy cladding tubes becomes smaller the higher the heat transfer from the cladding tube to the coolant is. This is attributable to tube bending occurring as a result of azimuthal differences in cladding tube temperatures and external cooling (see fig. 4). As the hot cladding tube side contacts more or less closely the heat source and the opposite cold side bends continuously off the inner heat source, intensified external cooling gives rise to an enhancement of the differences of the azimuthal cladding tube temperatures and, as a result, to a reduction in burst circumferential strain.

Fig. 7 makes evident that bundle tests which are performed with very low heat transfer, for instance low

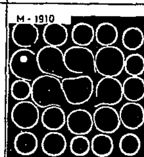
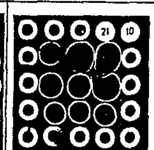
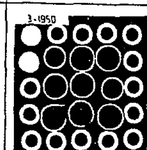
	REBEKA-M	REBEKA-2	REBEKA-3
cross-section at max. flow blockage			
fluid flow	stagnant steam	steam flow	two-phase flow
heat transfer coefficient [W/m^2·K]	<10	~30	~30 ÷ 100
mean burst strain of inner 3×3 rods [%]	63	54	44

Fig. 7. Influence of heat transfer on Zircaloy cladding deformation.

steam cooling, necessarily will lead to relatively great burst circumferential strains, whereas bundle tests in which heat transfer coefficients greater than 50 W/m^2 K dominate which are typical of the flooding phase of a loss-of-coolant accident yield relatively low burst circumferential strains.

In all experiments performed under typical heat transfer conditions average differences in the azimuthal cladding tube temperatures of about 30 K developed at the time of burst which limit the mean burst circumferential strain to values of approx. 50%.

3.2. Influence of the flow direction on coolant channel blockage

The coolant channel blockage caused by ballooned and burst cladding tubes in the fuel element depends, besides on the maximum circumferential strain of the deformed cladding tubes, also on the axial displacement of the burst points between the spacers. If the burst points are displaced over a rather large axial zone, the coolant channel blockage is relatively low, but if the burst points occur rather closely to each other, the resulting coolant channel blockage is greater for the same mean burst circumferential strain.

As plastic deformation of Zircaloy cladding tubes responds very sensitively to the cladding tube temperature, the axial displacement of the burst points is determined crucially by the axial profile of the cladding tube temperature of the individual fuel rods at the moment of failure and by its temperature maximum between two spacers. The cladding tube temperature profile inter alia is the result of the thermodynamic non-equilibrium in two-phase flow and its being influenced by the spacer grids.

The heat transfer between the rods and the mixture of steam and water droplets is achieved almost exclusively by convection. As the heat flow from the cladding tube wall to the steam is much stronger than the heat flow from the steam to the water droplets, a thermodynamic non-equilibrium develops during the flooding phase in two-phase flow which means that the steam is superheated along the coolant channel. In the bundle tests steam temperatures of up to about 600°C were measured which corresponds to about 450 K superheat.

At the spacer straps the incident water droplets are split up into smaller droplets so that on account of the greater droplet surface a more effective heat sink is produced for the highly superheated steam. Together with the enhanced turbulence downstream of each spacer this leads to a reduction in steam and cladding tube temperatures. However, up to the next spacer in the direction of flow, the degree of superheat increases again which leads to the development of an axial temperature profile and a temperature maximum between two spacers.

The direction of flow in the reactor core during a loss-of-coolant accident depends on the design and availability of the emergency core cooling systems and on their interaction with the primary circuits. Besides local differences in flow and steam/water counterflows, two characteristic and limiting flow directions exist in the reactor core as regards cladding tube deformation and coolant channel blockage in a combined injection mode into the cold and hot legs: flow reversal from the refill to the reflooding phases and unidirected flow during the refill and reflooding phases.

Fig. 8 illustrates the impacts of a flow reversal on the circumferential strain of the Zircaloy cladding tubes and the resulting coolant channel blockage. In the experiment (REBEKA 5) the rod bundle was passed by steam flow from top to bottom during the refill phase

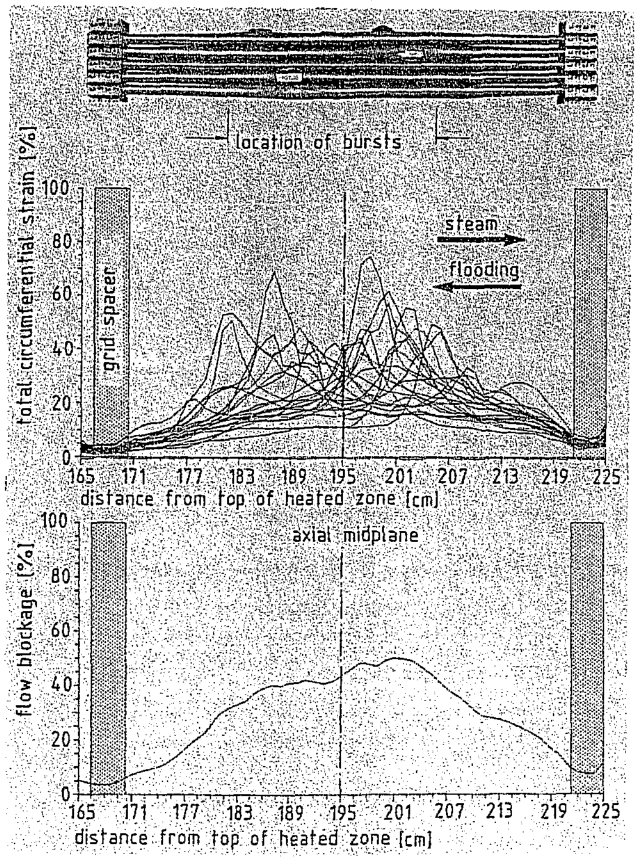


Fig. 8. Zircaloy cladding deformation and coolant channel blockage under reversed flow (REBEKA 5).

and from bottom to top during subsequent reflooding with water in order to simulate flow reversal. So, during the refill phase the cladding tube temperature maximum initially moves downward towards the spacer provided below the midplane as a result of the downward directed steam flow. In the subsequent flooding phase the temperature maximum is displaced in the direction of flow with the flooding time getting longer, towards the spacer provided above midplane, i.e., the temperature maximum between the spacers at different times occurs at different axial positions. But due to inhomogeneities in the rod bundle resulting from locally differing rod powers and cooling, not all the rods are heated up uniformly which gives different burst times. In REBEKA 5 the burst time interval of the individual Zircaloy claddings was about 24 s. During this time interval there was a shift in the temperature maximum which automatically led to an axial displacement of the burst points over a rather large range. It is evident from the figure that the burst points are spread over some 24 cm of axial length around the midplane which gives rise to a relatively low maximum coolant channel blockage of 52%.

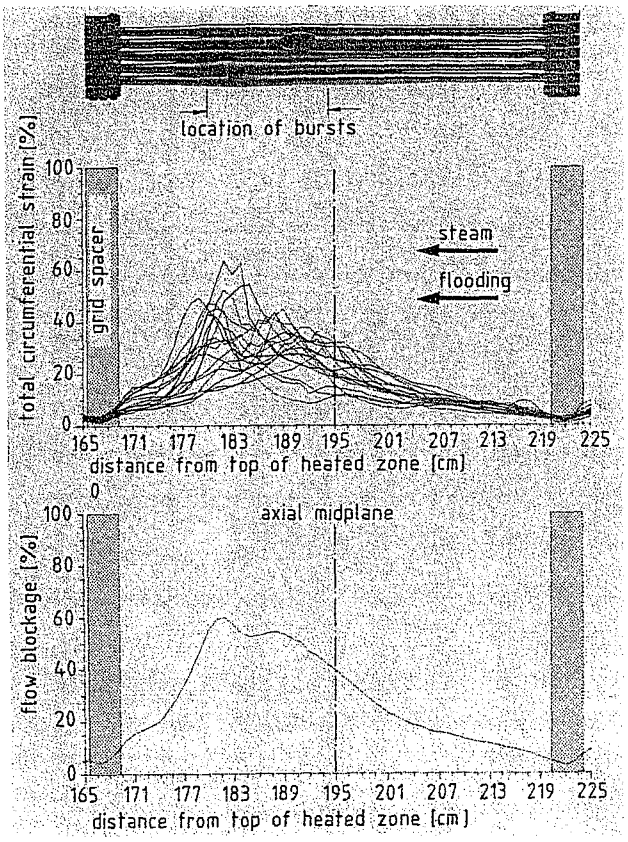


Fig. 9. Zircaloy cladding deformation and coolant channel blockage under unidirected flow (REBEKA 6).

Fig. 9 shows the deformation pattern for unidirected flow in the rod bundle. In this experiment (REBEKA 6) the flow direction of the coolant from bottom to top was maintained during the refill and reflooding phases. Unlike the REBEKA 5, the temperature maximum was moved from the very beginning of the experiment towards the upper of the two medium spacers. After this temperature profile had developed during the refill phase, the temperature maximum continued to occur at approximately the same axial positions. This leads automatically to a local concentration of the burst points and, consequently, to a stronger coolant channel blockage. The figure exhibits a pronounced displacement of the burst points in the direction of flow towards the upper spacer and illustrates the small cladding tube strains at spacers. The burst points are displaced only over an axial zone of about 14 cm because the flow direction has been maintained which results in a greater coolant channel blockage of 60%.

In the REBEKA 7 experiment the flow direction was likewise maintained, but the cooling conditions during flooding were set in such a manner that a maximum coolant channel could be expected. In this test the greatest coolant channel blockage to be expected under representative flooding conditions was approx. 70%.

4. Coolability of deformed rod bundles

The coolant channel blockage caused by ballooned cladding tubes in a rod bundle changes the cooling mechanism and induces two counteracting effects on the local heat transfer:

- Effect of lateral bypass flow of the blockage: This reduces the mass flow through the blocked zone and diminishes the heat transfer.
- Effect of passage through the blockage: This causes droplet atomization, flow acceleration and turbulence intensification and increases heat transfer.

In the FEBA program [8] forced flooding experiments were performed on a 5 × 5 rod bundle. Ballooned cladding tubes were simulated by conical sleeves fixed to electric heater rods. In the blocked area blockages of 62% and 90% were realized.

It was found that with a 62% blockage the effect of water droplet atomization by which heat transfer is improved dominates so that the cladding tube temperature is even lower in the blocked area than in the unblocked area. Fig. 10 shows the values measured for a 90% blockage. Under these extreme conditions the ef-

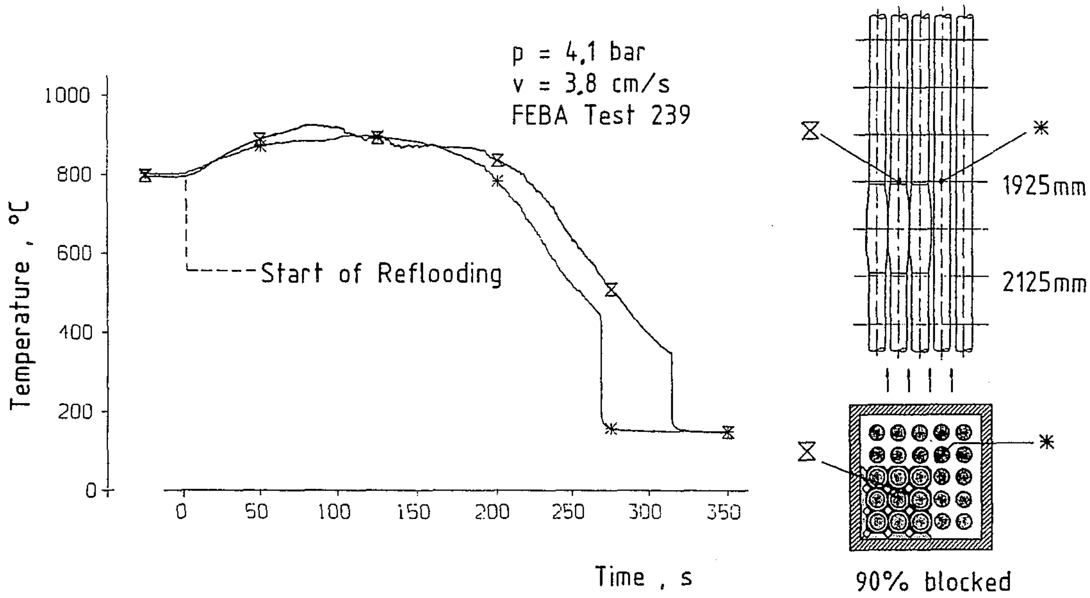


Fig. 10. Cladding temperatures in the 90% partly blocked rod bundle.

fect of lateral bypass flow of the blockage is dominating. Still, the temperature rise in the blocked zone and the extension of the rewetting period are insignificant.

This allows the conclusion to be drawn that the coolability in deformed fuel elements can be maintained up to coolant channel blockages of about 90%. Moreover, it has been proved in the REBEKA program that burst cladding tubes improve the coolability even further [9].

5. Summary and conclusion

Work performed on cladding tube deformation and core emergency cooling has provided sufficiently validated knowledge of the major mechanisms so that the safety of a pressurized water reactor can be assessed. Partial aspects which are still unanswered do not put in doubt the results obtained and their application in the licensing procedure.

The most important results can be summarized as follows:

- The number of the burst cladding tubes and their burst circumferential strain can be determined with sufficient accuracy if the temperature and pressure development of the fuel rods is known.
- The cooling effect of the two-phase flow which is intensified during flooding increases the temperature differences on the cladding tube circumference and limits in this way the mean burst circumferential strains to values of about 50%.
- A unidirected flow during the refill and reflooding phases leads to the greatest possible coolant channel blockage of about 70%.
- The coolability of deformed fuel elements can be maintained up to a coolant channel blockage of about 90%.

All effects described underline that in a loss-of-coolant accident no impairment whatsoever must be expected of the coolability of the fuel elements and that the safety margin applied in assessing the coolability is greater than predicted by most of the computer codes.

Therefore, it can be assumed that the safety of the population is fully guaranteed in the event of a loss-of-coolant accident.

Acknowledgement

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EXHIBIT C

ANSI/ANS-5.1-1994

American Nuclear Society

decay heat power in light water reactors

an American National Standard

published by the
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555 North Kensington Avenue
La Grange Park, Illinois 60525 USA

ANSI/ANS-5.1-1994

American National Standard for Decay Heat Power in Light Water Reactors

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-5.1**

Published by the
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555 North Kensington Avenue
La Grange Park, Illinois 60525 USA**

Approved August 23, 1994
by the
American National Standards Institute, Inc.

Table 5. Tabular data for standard decay heat power for thermal fission of ^{235}U following an irradiation of 10^{13} seconds

Time after shutdown t(s)	Decay heat power F(t,∞) (MeV/fission)[a]	One sigma uncertainty ΔF(t,∞) (MeV/fission)	One sigma uncertainty (percent)
1.0E+00	1.238E+01[b]	0.035E+01	2.8
1.5E+00	1.201E+01	0.030E+01	2.5
2.0E+00	1.170E+01	0.028E+01	2.4
4.0E+00	1.084E+01	0.024E+01	2.2
6.0E+00	1.026E+01	0.022E+01	2.1
8.0E+00	9.834E+00	0.197E+00	2.0
1.0E+01	9.497E+00	0.190E+00	2.0
1.5E+01	8.886E+00	0.169E+00	1.9
2.0E+01	8.460E+00	0.161E+00	1.9
4.0E+01	7.463E+00	0.134E+00	1.8
6.0E+01	6.892E+00	0.124E+00	1.8
8.0E+01	6.497E+00	0.117E+00	1.8
1.0E+02	6.202E+00	0.112E+00	1.8
1.5E+02	5.700E+00	0.103E+00	1.8
2.0E+02	5.373E+00	0.097E+00	1.8
4.0E+02	4.671E+00	0.084E+00	1.8
6.0E+02	4.287E+00	0.077E+00	1.8
8.0E+02	4.013E+00	0.072E+00	1.8
1.0E+03	3.799E+00	0.068E+00	1.8
1.5E+03	3.411E+00	0.061E+00	1.8
2.0E+03	3.140E+00	0.057E+00	1.8
4.0E+03	2.538E+00	0.046E+00	1.8
6.0E+03	2.238E+00	0.038E+00	1.7
8.0E+03	2.048E+00	0.035E+00	1.7
1.0E+04	1.912E+00	0.033E+00	1.7
1.5E+04	1.688E+00	0.030E+00	1.8
2.0E+04	1.549E+00	0.028E+00	1.8
4.0E+04	1.262E+00	0.023E+00	1.8
6.0E+04	1.121E+00	0.021E+00	1.9
8.0E+04	1.033E+00	0.021E+00	2.0
1.0E+05	9.729E-01	0.195E-01	2.0
1.5E+05	8.772E-01	0.175E-01	2.0
2.0E+05	8.191E-01	0.164E-01	2.0
4.0E+05	7.012E-01	0.140E-01	2.0
6.0E+05	6.368E-01	0.127E-01	2.0
8.0E+05	5.906E-01	0.118E-01	2.0
1.0E+06	5.547E-01	0.111E-01	2.0
1.5E+06	4.904E-01	0.098E-01	2.0
2.0E+06	4.463E-01	0.089E-01	2.0
4.0E+06	3.494E-01	0.073E-01	2.1
6.0E+06	3.020E-01	0.064E-01	2.1
8.0E+06	2.717E-01	0.057E-01	2.1

Table 6. Tabular data for standard decay heat power for thermal fission of ^{239}Pu following an irradiation of 10^{13} seconds

Time after shutdown (sec)	Decay heat power F(t,∞) (MeV/fission)[a]	One sigma uncertainty ΔF(t,∞) (MeV/fission)	One sigma uncertainty (percent)
1.0E+00	1.027E+01[b]	0.046E+01	4.5
1.5E+00	1.003E+01	0.042E+01	4.2
2.0E+00	9.822E+00	0.393E+00	4.0
4.0E+00	9.213E+00	0.350E+00	3.8
6.0E+00	8.802E+00	0.326E+00	3.7
8.0E+00	8.494E+00	0.314E+00	3.7
1.0E+01	8.250E+00	0.297E+00	3.6
1.5E+01	7.801E+00	0.281E+00	3.6
2.0E+01	7.483E+00	0.269E+00	3.6
4.0E+01	6.713E+00	0.242E+00	3.6
6.0E+01	6.257E+00	0.225E+00	3.6
8.0E+01	5.935E+00	0.214E+00	3.6
1.0E+02	5.691E+00	0.205E+00	3.6
1.5E+02	5.268E+00	0.190E+00	3.6
2.0E+02	4.988E+00	0.180E+00	3.6
4.0E+02	4.363E+00	0.157E+00	3.6
6.0E+02	3.999E+00	0.148E+00	3.7
8.0E+02	3.733E+00	0.138E+00	3.7
1.0E+03	3.522E+00	0.130E+00	3.7
1.5E+03	3.134E+00	0.119E+00	3.8
2.0E+03	2.864E+00	0.112E+00	3.9
4.0E+03	2.282E+00	0.094E+00	4.1
6.0E+03	2.009E+00	0.084E+00	4.2
8.0E+03	1.845E+00	0.081E+00	4.4
1.0E+04	1.733E+00	0.076E+00	4.4
1.5E+04	1.554E+00	0.071E+00	4.6
2.0E+04	1.443E+00	0.068E+00	4.7
4.0E+04	1.211E+00	0.058E+00	4.8
6.0E+04	1.087E+00	0.053E+00	4.9
8.0E+04	1.006E+00	0.050E+00	5.0
1.0E+05	9.482E-01	0.474E-01	5.0
1.5E+05	8.541E-01	0.427E-01	5.0
2.0E+05	7.951E-01	0.406E-01	5.1
4.0E+05	6.695E-01	0.341E-01	5.1
6.0E+05	6.005E-01	0.306E-01	5.1
8.0E+05	5.523E-01	0.282E-01	5.1
1.0E+06	5.157E-01	0.263E-01	5.1
1.5E+06	4.525E-01	0.231E-01	5.1
2.0E+06	4.107E-01	0.209E-01	5.1
4.0E+06	3.223E-01	0.168E-01	5.2
6.0E+06	2.802E-01	0.146E-01	5.2
8.0E+06	2.537E-01	0.132E-01	5.2