UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING AND POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit 1)	§	

APPLICANT'S RESPONSE TO "JOHN F. DOHERTY'S AMENDED CONTENTIONS NUMBERED 4, 11, 15, 16, 20, 21, 36, 38 and 39 .

Houston Lighting & Power Company (Applicant) hereby submits the following individual responses to the amended contentions filed by John F. Doherty (Intervenor).

Amended Contention 4. ATWS.

In this second amendment to his contention on ATWS, Intervenor has again failed to present a litigable issue. Applicant's original response to this contention stated clearly that "...the Applicant must, and indeed is willing to stipulate, that it will comply with whatever NRC requirements are ultimately established with respect to the ATWS generic issue." Intervenor has presented nothing which might further suggest that Applicant's ATWS commitment is inadequate.

Moreover, there is nothing in Intervenor's list of modifications requiring "flexibility of design"

which suggests that whatever changes are necessary to fully implement the generic resolution cannot be accommodated long before construction on the pertinent components and structure begins.[1/] Without this assertion and some reasonable basis for it, no contention exists.

Amended Contention 11. Spent Fuel Pool Loss of Water.

Intervenor contends that the spent fuel pools in the fuel handling building and containment building[2/] are subject to a complete loss of all water accident, which could result in melting of fuel rods and subsequent radiation release. The only support given for this postulation is a government report entitled, "Spent Fuel Heatup Following Loss of Water During Storage", NUREG/CR-0649, Allan S. Benjamin, March, 1979. In response to this exact same contention, offered by petitioners Madeline and Robert Framson,

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1/ Only two items on Intervenor's list -- (b) and (g) -- are actual unincorporated changes required by the alternative systems discussed in NUREG/CR-0460. Three have nothing to do with ATWS whatsoever -- (e), (f), and (i); and four are already capable of accommodation in the ACNGS design -- (a), (c), (d), and (h).

2/ "Spent fuel can be stored both in the Reactor Building and in the Fuel Handling Building. However, fuel will not be stored in the Reactor Building except during periods of refueling, on a temporary basis." ACNGS PSAR, Section 9.1.2.2.

both Applicant and Staff noted that there is absolutely nothing in the contention demonstrating any relationship between NUREG/CR-0649 and the spent fuel design for ACNGS. In fact, NUREG/CR-0649 deals only with the consequences to stored spent fuel after a postulated complete drainage of the storage pool -- an event whose likelihood the report concludes "is judged to be extremely low." (p. 12) Furthermore, the report notes specifically that "(a)ccident initiation mechanisms the probability of occurrence, the magnitude of radioactive release, or the public consequences are not addressed." (p. 11). Hence, NUREG/CR-0649 does not support any allegation that a spent fuel pool loss of water accident is a probable event, a dangerous event, or an event which threatens public health and safety. Consequently, the contention should be rejected.

Amendment to Contention 15. WIGLE power excursion theory.

Intervenor argues that the reactivity model WIGLE does not properly account for a "rapid increase in reactivity." Applicant has twice responded that WIGLE is not used to analyze any event for ACNGS. In an attempt to side-step this clear rebuttal, Intervenor now states that the "one dimensional calculation of the scram reactivity" function that is used by Applicant does not compare with "data

resulting from the neutron burst experiments reported in IN-1370." The assertion is totally irrelevant because the "one dimensional calculation of scram reactivity" referenced on p. 4-11 of ACNGS SER Supplement 2 is used for calculating the operating transient scram reactivity, not for calculating the negative scram reactivity in events which produce a "rapid increase in reactivity."[3/] Intervenor's contention that the one-dimensional reactivity function is used for evaluating events resulting in rapid reactivity insertion is clearly contradicted in NEDO-10527, which states: "The primary design method at General Electric for analysis of super-prompt critcal large-core nuclear excursions uses the adiabatic approximation with a two-dimension multi-group flux calculation." Hence, since Intervneor has confused

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3/ The worst case example of an event which produces rapid reactivity increases is the rod drop accident. The difference in the operating scram reactivity and the rod drop reactivity calculations are described by the ACNGS PSAR, p. 4.3-26. "The total scram reactivity and scram functions which are used for analyzing the rod drop excursions obviously are quite different in nature than those used for the operating plant transients since the most severe rod drop accidents occur in startup and low power ranges where the void distribution is either nonexistent or very dissimilar and the control rod patterns vary greatly from those observed at operating conditions. As was the case for the scram characteristics for the plant operational transients, the scram characteristics used for analyzing the rod drop accident are strongly dependent on the fuel design and fuel loading pattern."

reactivity calculations used during normal operation with those used for particular accident conditions, the contention has no real basis and should be denied.

Amendment to Contention 16. Steam blanketing of blocked fuel assemblies.

In his May 25, 1979 contentions, Intervenor contended that "fuel rods will cause steam blanketing of the Emergency Core Cooling System (ECCS) coolant" (Contention 16, May 25 Contentions, at 6). Applicant opposed this contention as either an impermissible challenge to the Commission's ECCS regulations or a nonspecific complaint about Applicant's compliance therewith. Intervenor separately contended that

> "[t]he design based [sic] accident for a flow blockage incident is inadequate because it assumes blockage of but one fueld assembly." (Contention 25, May 25, Contentions at 10.)

In its response, Applicant stated that Contention 25 presented a litigable issue.

Now Intervenor in his amendment to Contention 16 seeks to bootstrap Applicant's acquiescence with Contention 25 into a litigable amended Contention 16 by asserting that steam blanketing will result from flow blockage caused by "an object in the reactor working loose." This attempt fails.

1228 314

Intervenor does not state why steam blanketing is a probable, or even possible, result of this postulated loose object in the core[4/]. Intervenor has provided no information, either experimental or theoretical, suggesting that the steam blanketing phenomenon exists, much less whether it should be accounted for in the flow blockage analysis. Without some arguable support for the connection between steam blanketing and flow blockage, stated with reasonable specificity, there is no basis to the new contention and it should be dismissed.

Amended Contention 20. Gap conductance.

Intervenor has, by amending this contention further clouded an already indecipherable dissertation. Intervenor raises a new allegation that fission gas release due to fuel rupture during a LOCA will result in lower pellet clad gap conductance. Applicant is at a total loss to understand how a lower gap conductance can result if a rod has ruptured, thereby releasing its fission gases. The rest of the amendment makes no sense in light of this basic misconception.

4/ Intervenor's reference to the incident at Fermi 1 is certainly not helpful, since that reactor was a sodium-cooled breeder reactor and could not possibly have had steam blanketing.

The only intelligible clue to whatever issues Intervenor may have in mind is a reference to an article in Nuclear Safety; Volume 20, Number 4; July - Aug., 1979; p. 418. The "article" is a letter to the editor from Howard Ocken and J. T. A. Roberts of the Electric Power Research Institute suggesting that fission gas release rates for LWR fuel rods are more dependent on temperature than fuel burnup. The letter is followed by an answer from the Commission Staff acknowledging the dominant influence and stating that its analysis accounts for temperature sensitivity. Nothing in the letter or reply correlates with the confusing allegations and assertions contained in Intervenor's "contention."[5/] In sum, this amendment lacks coherency and basis. It should be dismissed.

Amendment to Contention 21. Void collapse reactivity.

This contention relies solely on unsupported speculation. Intervenor claims future information may show that the reactivity inserted by the collapse of voids during overpressure transients is underestimated. This result may cause the NRC staff to impose technical specification restrictions on Allens Creek which may result in derating of the

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5/ The letter and reply are attached to this response.

1228 316

plant. This might affect Intervenor's environmental interests if more environmentally unacceptable means are used to generate electricity. Not once in this dependent series of hypothesis does Intervenor supply any reason to believe that any of this conjecture will materialize, let alone the postulated "domino" result. Obviously such a string of speculations does not form the basis for a litigable contention.

Amended Contention 36. Drywell vacuum breaker.

Intervenor's first complaint is that the drywell vacuum breaker sizing problem "is not reported resolved 10 months after the first special pre-hearing conference." However, Applicant directs Intervenor's attention to page 6-3 of the ACNGS SER Supplement 2 (March, 1979) where the NRC Staff states that "we conclude that the drywell vacuum relief system design is acceptable."

Intervenor then discusses bypass leakage of vacuum breakers in the drywell wall during an inadvertent starting of the containment spray system (CSS). The Staff postulated inadvertent CSS starting as a design parameter for the containment vacuum relief system, not the drywell vacuum breakers. In this regard, inadvertent CSS start analysis has been satisfactorily accomplished for ACNGS. See SER Supplement 2, pp. 6-3 and 4.

Finally, Intervenor alleges that mass transfer effects were not accounted for in the analysis submitted by Applicant of the inadvertent operation of containment spray. Again, this issue is relevant only to containment vacuum breakers. In any event, at page 6-4 of SER, Supplement 2 (p. 6-4) the Staff reports that:

> The effects of mass transfer to the sizing of containment vacuum breakers has been considered in our review of the GESSAR-238 Nuclear Island. As reported in "Safety Evaluation Report as Related to the Preliminary Design of the GESCAR-238 Nuclear Island Standard Design General Electric Company, Supplement No. 2" NUREG-0124 (Supp. 2 to NUREG 75/110) January 1977, we found the method of analysis acceptable provided that, during normal plant operation, the containment temperature and relative humidity are maintained within containment temperature limits used in the General Electric Company's vacuum breaker spray analysis. The applicant has stated that Allens Creek design will conform to this resolution.

All three of Intervenor's complaints are plainly without basis; accordingly, the contention and amendment should be rejected.

Contention 38. RHR single failure proof.

Both Applicant and Staff opposed the July 31, 1979 filing of amended contention #38 on the grounds that Intervenor failed to present a litigable issue regarding compliance with NRC regulations and further failed to specify what "systems interaction" issues are pertinent to the ACNGS RHR system. Intervenor has failed to cure these defects in his

1228 318

latest filing, excepting that he has deleted all discussion of systems interaction.

Intervenor again asserts that Applicant does not comply with GDC 19 and 34 with regard to "...bringing the reactor to cold shutdown in 24 hours" and references NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations" in support of this assertion. This reliance on NUREG-0578, however, is misplaced. Item 2.2.3 of NUREG-0578 recommends revisions to Technical Specifications which would require that "...the reactor be placed in a hot shutdown condition within 8 hours and in a cold shutdown condition by the licensee within 24 hours of any time that it is found to be or to have been in operation with a complete loss of safety function...". This recommendation in NUREG-0578 is a proposal for an administrative penalty upon loss of a safety system (NUREG-0578 at A-63). As such, it is not at all relevant to Intervenor's allegation that the RHR system fails to conform to GDC 19 and 34, nor is it relevant to Intervenor's incorrect assertion that GDC 19 and 34 require the RHR system must be capable of bringing the reactor to cold shutdown in 24 hours. The NRC Staff, in response to Intervenor's July 31, 1979 amendment to Contention 38, stated that: "GDC 19 and 34 do not specify any period of time required for cold shutdown capability" and that

"...there is no basis for this part of the contention...". (NRC Staff Response to Contention #40 and Amendments to Previously Submitted Contentions of John F. Doherty, August 14, 1979). Applicant submits that this objection still holds.

Moreover, Applicant would note that Section 5.5.7.2 of the ACNGS PSAR clearly states the RHR system will have enough heat removal to cool down the reactor to 125° F within approximately [illegible] hours after shutdown. Thus, even if Intervenor's addi[illegible] GDC 19 and 34 was correct, Applicant would comply. Hence, there certainly is no litigable issue.

Finally, Intervenor again proposes a remedy to his concern in suggesting that Applicant be required to meet the requirements of NUREG-0152 (page 5-21) and NUREG-0190 Appendix A (which are the same document). Applicant repeats that its design does meet the requirements of NUREG-0152 page 5-21. See Section 5.4.5 of Supplement 2 to the Allen's Creek SER. Thus, Contention No. 38 should be rejected as toally lacking basis.

Contention #39. Fuel rod ballooning.

In this amendment to a previous untimely contention, Intervenor substantially repeats himself. Both Staff and Applicant opposed this contention previously on the grounds that Intervenor had failed to establish a clear nexus between the fuel ballooning in TMI-2 accident and the possibility of

similar occurrences at ACNGS. Although Intervenor alleges some arbitrarily chosen dimensional and material similarities between Allen's Creek and TMI-2 fuel, he fails to specify the relevance of the supposed similarities to the fuel rod ballooning issue.

As part of amended contention 39, Intervenor now asserts as another untimely contention that Applicant's fuel rods are not in compliance with 10 CFR Part 50, Appendix K. Intervenor fails to specify in any reasonably specific manner why this is true, or to specify in what way the ECCS analyses in the ACNGS PSAR might be in error. It may be that Intervenor challenges the ECCS criteria on the ground that they do not adequately consider fuel ballooning. That is an error but, in any event, Intervenor does not begin to establish the bases for an attack on Commission regulations.

Intervenor has again relied heavily upon NUREG-0557 and ORNL-4752 as a basis for this contention. The Staff concluded in response to Intervenor's previous filing that "moreover, since NUREG-0557 represents both Mr. Doherty's excuse for the late filing and the substantive basis for the contention, the contention must be rejected regardless of lateness because it lacks the nexus between TMI and Allen's Creek necessary to supply the basis required by 10 CFR 2.714." The staff also states that "...with regard to

ORNL-4756, Mr. Doherty has made no attempt whatever to relate the test conditions at Oak Ridge to the Allens Creek design of 8 x 8 fuel." Applicant is in full agreement with these two dispositive objections.

Intervenor has failed to cure the several inadequacies in this untimely contention and, thus, it should be dismissed.

Respectfully submitted,

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1228 322

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CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Applicant's Response to John F. Doherty's Amended Contentions Numbered 4, 11, 15, 20, 21, 36, 38 and 39 in the above-captioned proceeding were served on the following by deposit in the United States mail, postage prepaid, or by hand-delivery this 28th day of September, 1979.

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1228 324

Table 3 Probabilities of Eventual Death from Different Competing Risks

	All risks except nuclear	All risks except nuclear and motor vehicle	All risks	
			$\lambda = 2 \times 10^{-10}$	$\lambda = 10^{-6}$
Other	0.976	1.0	0.976	0.976
Motor vehicle	0.024	0.0	0.024	0.024
Nuclear	0.0	0.0	10^{-6}	7×10^{-5}
Life expectancy, years	72.8	73.6	72.8(−18 s)	72.8(−1 d)

quantifying this kind of risk. The above computations are deficient, however, since many fatalities due to nuclear accidents will not be immediate. Further work is needed to quantify the risk of these delayed fatalities.

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LETTER TO THE EDITOR: COMMENTS ON "FISSION-GAS RELEASE FROM FUEL AT HIGH BURNUP" IN VOL. 19, NO. 6

Meyer, Beyer, and Voglewede[1] have proposed that an enhancement factor be applied to existing vendor models when fission-gas release (FGR) at burnups greater than 20,000 MWd/metric ton is calculated for licensing purposes. This enhancement factor is derived from FGR data obtained from liquid-metal-cooled fast breeder reactor (LMFBR) fuel. The analysis assumes that the intrinsic source of the high FGR measured for some light-water-reactor (LWR) fuel rods is, per se, the burnup. In deriving the enhancement factor, the authors claim that effects on FGR due to differences in operating temperature between LWR and LMFBR fuels have been taken into account.

The following comments are submitted to suggest that an alternative factor, namely, fuel operating temperature, is the dominant variable that determines FGR. Recent FGR data from LWR fuel rods that support this view are presented. It is argued that temperature effects can be used to rationalize the high FGR values from LWR fuel rods that were attributed in Ref. 1 to high burnups. This note also questions whether the proposed enhancement factor truly accounts for known differences in operating temperatures between LWR and LMFBR fuels.

Gaseous fission products form directly from the fissioning of ^{235}U and ^{239}Pu or by the radioactive decay of other fission products. Higher burnup, per se, only increases the inventory of fission gas that potentially can be released. Fission-gas release requires that the gas atoms, which have been generated in the interior of the fuel, reach a free surface of the fuel. Various mechanisms have been proposed to account for the release of fission gases from fuel. In the range of interest (i.e., where FGR is on the order of a few percent), the majority of investigators support the view that migration of fission gases in the form of atoms or

bubbles to a surface that can communicate with the free space of the fuel rod is required. Such migration occurs by a thermally activated diffusion process that obeys an Arrhenius equation, and this suggests that temperature plays the central role in the FGR process.

In this light it is instructive to review the methodology used to derive the enhancement factor of Ref. 1. The enhancement factor is an exponential function containing three arbitrary constants. The sole independent variable is burnup. High-burnup LMFBR FGR data obtained by Dutt and Baker[2] served as the basis for evaluating these constants. In calculating FGR at high burnups, an existing vendor model is first used to calculate FGR at 20,000 MWd/metric ton. This calculated value is then weighted by the enhancement factor, which is evaluated at the burnup of interest. Since burnup is the only independent variable defined by the enhancement factor, effects of operating temperature on FGR have not been explicitly taken into account in deriving this factor. Since, for the same rating, LMFBR operating temperatures are higher than LWR operating temperatures, the enhancement factor is conservative with respect to the predicted FGR from LWR fuel rods. The enhancement factor is not imposed on low-burnup (<20,000 MWd/metric ton) FGR calculations, it being argued that current FGR models are adequate in this range. Some degree of conservatism is assured, however, by requiring that a minimum value of 1% be used for the FGR at 20,000 MWd/metric ton.

Fission-gas-release data that bear on this issue have been obtained from projects sponsored by the Electric Power Research Institute (EPRI). Measurements have been obtained from fuel rods irradiated in seven commercial LWRs, and additional measurements are to be obtained from two other reactors. The data, together with analyses using the COMETHE-IIIK computer program, support the view that fuel operating temperature is the key variable that determines FGR. The role of burnup is secondary compared with that of fuel operating temperature.

Figure 1 presents the results of the EPRI-sponsored FGR measurements as a function of burnup.[3] Significant FGR (>1%) is not restricted to high-burnup values. FGR values ranging from 5.7 to 15.3% were observed in 12 rods with burnups less than 20,000 MWd/metric ton. At higher burnups, from 20,000 to 30,000 MWd/metric ton, FGR values ranging from 6.9 to 24.2% were observed in 26 rods. The factor common to these rods was that they were unpressurized. This appears to be a necessary, but not sufficient, condition for high FGR since small FGR values have been reported for other unpressurized rods at burnups to 24,000 MWd/metric ton. Small FGR (<1%) was observed in all pressurized rods.

The results indicate that for these unpressurized rods a transition from low FGR values to high FGR values occurs over a very narrow burnup range. This transition region, or threshold for significant FGR, occurs at about 12,000 MWd/metric ton for the Maine Yankee fuel rods and at about 24,000 MWd/metric ton for the Oyster Creek fuel rods. Such a FGR threshold value also argues against the burnup dependence of FGR as proposed in Ref. 1. Comparison of FGR data from sibling rods irradiated in the Big Rock Point reactor[4] also supports the idea of a threshold value for significant FGR. Data for three pairs of rods (Table 1) show that, for modest differences in burnup of about 15%, FGR values differing by more than a factor of 10 are observed. Phenomenological models of FGR that predict the general trends observed in the Oyster Creek, Maine Yankee, and Big Rock rods (i.e., nearly zero FGR up to a threshold burnup value, followed by a transition to high FGR values over a narrow burnup range) have been reported by Dollins and Nichols[5] and by Hargreaves and Collins.[6]

The COMETHE-IIIK fuel performance computer program is being used, together with appropriate duty cycle and fuel fabrication data, to interpret these FGR data.[7] These analyses support the view that fuel-rod operating temperature is the key determinant of FGR. The factors that appear to be most important in establishing fuel temperatures are the linear heat rating, the fuel-clad gap size, and the thermal conductivity of the gap. The gap size, in turn, is strongly influenced by clad creepdown and fuel densification rates. The data in Fig. 1 suggest that gap conductivitiy in unpressurized rods can be degraded to such an

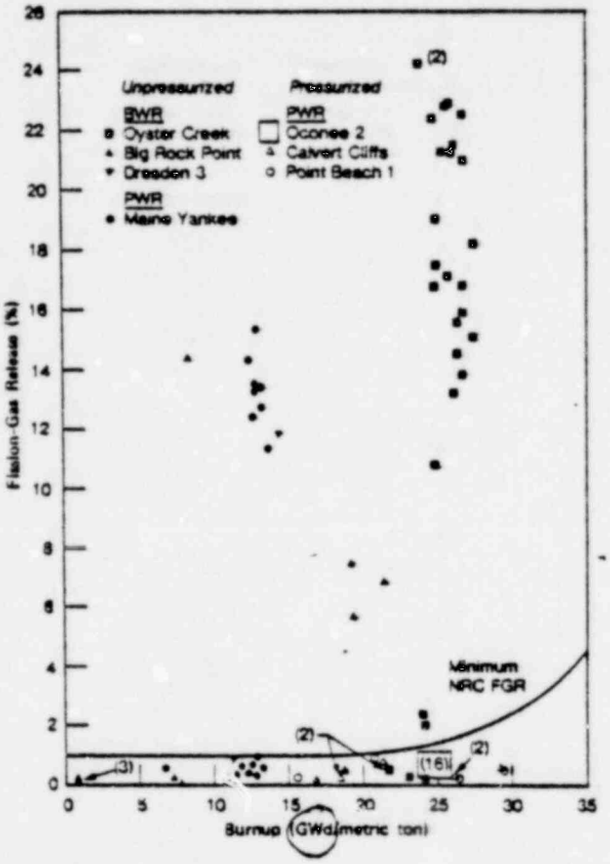

Fig. 1 Fission-gas release vs. burnup data obtained from LWR fuel rods. The minimum fission-gas release calculated with the NRC enhancement factor is also shown.

Table 1 Fission-Gas-Release Data from Fuel Rods Irradiated in Big Rock Point

Rod number	Average burnup, MWd/metric ton	Fission-gas release, %	Fuel-clad gap, mils	Pellet density, % theoretical density	First cycle fuel shrinkage, % ΔL/L	Peak pellet power by cycle, kW/ft 1st	2nd	3rd
JJ 400002	8,700	13.7	9.5	89.02	2.65	9.2		
JK 400001	8,000	0.3	9.5	90.12	2.32	8.2		
AB 30001	21,500	6.9	9–13	94.0	1.12	9.0	13.7	4.4
AB 20001	18,300	0.52	9–13	93.5	0.80	7.8	12.8	4.1
AB 10001*	19,200	7.5	9–13	92.2	1.52	7.2	13.0	4.3
AB 40001	16,800	0.21	9–13	93.75	1.73	6.8	10.8	3.6

*Gadolinium-bearing rod.

1228 327

extent that significant FGR will result. On the other hand, for these fuel rods, pressurization is effective in maintaining high values of thermal conductivity across the gap so that FGR is kept at values below the minimum 1% value required by Ref. 1 even at burnups approaching 30,000 MWd/metric ton.

If burnup is not the key determinant of FGR at high burnups, how can one rationalize the LWR data of Ref. 1 that were offered as evidence to support the FGR enhancement factor? Since FGR data are presented there to burnups approaching 60,000 MWd/metric ton, these data reflect the response of LWR fuel fabricated from the late 1960s to the early 1970s. During this period, effects resulting from irradiation-induced fuel densification were not recognized by the nuclear industry. Also, some of the data of Ref. 1 were obtained from fuel rods operated at high ratings, far in excess of that found in commercial LWRs.[8] We suggest that high FGR values were observed because high ratings and irradiation-induced densification led to operating temperatures in these fuel rods that are higher than would prevail in commercial LWR fuel rods. These high temperatures, in turn, led to high FGR values. In the interim period the factors responsible for irradiation-induced densification have been determined,[9] and the fuel vendors are routinely fabricating densification-resistant fuel. As this newer fuel achieves higher burnups, and as older fuel is discharged, it is likely that lower FGR values will be observed.

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AUTHORS' RESPONSE TO THE PRECEDING LETTER

We have received a copy of Ocken and Roberts' comments on Fission-Gas Release from Fuel at High Burnup, which was published in the November–December 1978 issue of *Nuclear Safety*. Our response to these comments is as follows.

1. Application of NRC correlation requires knowledge of both fuel burnup *and temperature* at and beyond 20,000 MWd/metric ton. We have not assumed that burnup is the dominant variable in the correlation. To the contrary, we believe that temperature is the strongest variable and therefore have retained the users temperature dependence *F(T)* at all burnups.

2. If high-temperature fission-gas release is a diffusion-controlled process, as Ocken and Roberts suggest, one would expect it to be both thermally activated and concentration dependent. The question is not, therefore, the existence of a burnup dependent effect, but its magnitude.

3. The authors further suggest that the range of interest for fission-gas release is on the order of a few percent. While this may be the case for low-burnup prepressurized fuel rods operated at nominal power levels, it is obviously not the case for all fuel designs, particularly those operated at high linear heat ratings of interest in licensing (i.e., the licensed LWR power limits of around 13 to 15 kW/ft).

Our original publication assumes the role of burnup with respect to temperature to be obvious. Perhaps we did not give this point enough emphasis. As evidenced by Ocken and Roberts' comments, this point does not appear to be clearly understood, and the potential for misapplying the NRC correlation exists. . . .

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Core Performance Branch
Division of Systems Safety
Nuclear Regulatory Commission
