

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 11, 1976

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

Dear Mr. Anders:

During its 190th meeting, February 5-7, 1976, and its 191st meeting, March 4-6, 1976, the Advisory Committee on Reactor Safeguards completed a review of the proposal to increase the maximum operating power level of the Donald C. Cook Nuclear Plant Unit No. 1 from 81% of rated power to the rated power of 3250 Mwt. The Committee has previously discussed this project in its reports of December 13, 1968, and October 17, 1973. A Subcommittee meeting on the current proposal was held in Washington, D. C., on February 4, 1976. During its review, the Committee had the benefit of discussions with representatives of the Indiana and Michigan Power Company, American Electric Power Service Corporation, Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

In its report of October 17, 1973, the ACRS recommended that a continuing series of monitoring and evaluation measures be applied to this first-of-a-kind ice condenser system employed as part of the containment for Donald C. Cook Nuclear Plant Unit No. 1. The Applicant has conducted frequent periodic surveillance, including sample weighing of ice baskets. The results of this surveillance program indicate a larger rate of ice loss than previously anticipated and a nonuniform rate of ice loss across ice bays and among ice bays. Baskets adjacent to walls appear to be subject to increased rates of ice loss; however, such baskets are relatively inaccessible for weighing, and efforts to date have not led to an accurate method for measuring their weights.

The Committee believes that the current inventory of ice is acceptable. However, the Committee believes that further analysis and evaluation are required to ascertain the minimum acceptable basket weights, particularly in wall baskets, and to establish sampling and measurement methods which are adequate to assure that these limits have been met. The ACRS believes that for the next year an appropriate sampling program, performed every three or four months as is currently planned by the Applicant, should provide suitable assurance of an adequate inventory while an improved basis for setting the inventory limits and sampling procedures is developed. The Committee wishes to be kept informed with regard to this matter.

In its report of October 17, 1973, the ACRS also recommended that the Regulatory Staff independently evaluate ice condenser behavior under postulated accident conditions. The NRC Staff reported that such confirmation has not yet been accomplished but that they expect to have such a capability within the year.

The Applicant reported the results of a program of verification of the adequacy of the axial power distribution monitoring system (APDMS) for steady state operation and for several transient power conditions. The experience has been favorable.

The Applicant reported that the LOCA-ECCS analysis, on the basis of the Westinghouse evaluation model approved in March, 1975, would limit the maximum allowable nuclear peaking factor (F_Q) to 1.84 without allowance for rod bowing at rated power. Westinghouse has recently proposed several changes in the evaluation model and, in a modified form, these have been tentatively accepted by the NRC Staff. When the Donald C. Cook Nuclear Plant Unit No. 1 is analyzed for operation covering only the remainder of the first fuel cycle, and when as-built parameters are used together with the modified changes in the evaluation model, a maximum F_Q of 1.98 (plus an additional factor of about 0.08 for bowing) is calculated to correspond to the limits of Appendix K of 10 CFR 50 at rated power.

In view of the favorable experience with APDMS and the expected operation of Donald C. Cook Nuclear Plant Unit No. 1 in a base loaded fashion for the remainder of the first fuel cycle, the Committee believes it acceptable for this plant to be operated at powers up to rated power for the first fuel cycle. This matter should be reviewed by the NRC Staff and the ACRS prior to operation at rated power with the next fuel load.

The ACRS reiterates its recommendations of October 17, 1973, that consideration be given to the possibility of improvements in ECCS effectiveness to provide additional margin.

The NRC Staff is evaluating, on a generic basis, the adequacy of reactor vessel support systems. Some aspects of reactor vessel cavity pressure loadings during postulated accidents may also warrant reexamination in connection with the application of the results of the generic study to Donald C. Cook Nuclear Plant Unit No. 1. This matter should be resolved in a manner satisfactory to the NRC Staff.

The Committee recommends that consideration of cumulative downtime of safety-related components and systems be included in the continuing development of technical specifications for Donald C. Cook Nuclear Plant Unit No. 1.

March 11, 1976

The Committee recommends that continuing evaluation be made of the possibility of further enhancing the provisions for industrial security.

The NRC Staff has not yet completed its generic evaluation of possible modifications in requirements for protection against fires in light water power reactors. The Committee wishes to be kept informed of the results of application of this study to Donald C. Cook Nuclear Plant Unit No. 1.

Generic problems relating to large water reactors are discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The ACRS believes that, subject to the foregoing and to matters discussed in its report of October 16, 1973, the Donald C. Cook Nuclear Plant Unit No. 1 can be operated at powers up to rated power for the first fuel cycle. This matter should be reviewed by the NRC Staff and the ACRS prior to operation at rated power with the next fuel load.

Additional comments by Drs. David Okrent and Milton Plesset, Dr. Herbert Isbin, and Mr. Myer Bender are presented on the following pages.

Sincerely yours,



Dade W. Moeller
Chairman

Additional Comments by Drs. David Okrent and Milton Plesset

For several reasons, we do not concur with the recommendation to permit Donald C. Cook Nuclear Plant Unit No. 1 to operate at rated power at this time.

First, while there may be merit in the proposed changes in the Westinghouse evaluation model, we believe further examination is warranted of several factors, including the scaling of experiments, the scatter in data, and the possible influence of super-plasticity on clad behavior during postulated loss-of-coolant accidents. Our reluctance to endorse these changes is also due, in large part, to signs of a continued process of cutting into the conservatism built into the original evaluation models, without a concomitant buildup in our basic understanding or predictive ability for the overall LOCA-ECCS process. In this situation there are limits beyond which the use of best estimate heat transfer coefficients, etc., is no longer appropriate.

Second, even with application of the revised Westinghouse evaluation model which has been judged acceptable by the NRC Staff, Donald C. Cook Nuclear Plant Unit No. 1 requires a LOCA - limited maximum peaking factor (F_0) of 1.98 (plus the margin for bowing) at rated power. While this is somewhat higher than the F_0 which can be expected at steady operation for the rest of the first fuel cycle for Donald C. Cook Nuclear Plant Unit No. 1, it still represents a very large reduction in the margin that has been available for most plants between LOCA - limited F_0 and that value which would be present most of the time. This margin has been eroded until it is a small fraction of its earlier values. Furthermore, if we accept this low F_0 value for Donald C. Cook Nuclear Plant Unit No. 1, a precedent will be set by means of which all PWR's will be able to reduce what was a substantial safety margin only a few years ago. This previously available substantial safety margin could cover many of the existing uncertainties in the analysis of LOCA-ECCS. The uncertainty aspect is highlighted by the less than perfect record obtained by the experts in their pre-prediction of various separate effects experiments, by the recognized difficulties in a calculation from first principles, by the current unavailability of experiments to test all relevant effects, and by the lack of a meaningful test of Westinghouse predictive capability with experiment.

Third, the ACRS has in the past been reluctant to accept proposed operation of reactors with F_0 's less than 2.2. In part, such caution arose from the knowledge that, with a more flattened power distribution, a much larger fraction of the fuel elements would be at or near peak temperatures, given a LOCA, and therefore potentially vulnerable to an "anomaly" in ECCS function (such as some three-dimensional flow effect or excessive steam generator leakage).

Fourth, we find an absence of effort on the part of the Applicant to develop possible improvements in ECCS capability and reliability and too small an effort by the nuclear industry or the NRC safety research program itself in this regard.

We believe that, while experience with the APDMS is favorable, it would be prudent to permit operation of Donald C. Cook Nuclear Plant Unit No. 1 at power levels only up to 90% or 92% of rated power during the remainder of the first fuel cycle. During this period, further information should become available on LOCA-ECCS, on the response of more highly flattened cores to anomalous ECCS function, and also on ice-condenser behavior. Equally important, a precedent for operation with such low F_0 values would not be set, and the industry might be encouraged to develop improvements in ECCS, particularly with regard to reflooding rates and an insensitivity to steam binding, which the ACRS in many reports and the AEC Commissioners, themselves, in their decision on the ECCS Acceptance Criteria, have urged.

Additional Comments by Dr. Herbert S. Isbin

Although I concur with my colleagues that the Applicant can operate the Donald C. Cook Nuclear Plant Unit No. 1 at full power and meet the requirements set forth by the NRC Staff's evaluation of the LOCA-ECCS analyses, I suggest restricting the operations to a limit less than 100% power. Furthermore, I believe that though this action may place an unusual burden upon this Applicant, the contributions to be made will, in the long range, advance safety in the nuclear industry. My recommendation is to arbitrarily restrict the power level to about 90% and is based upon the following factors:

1. The current operating limit is 81% and the proposed increase in power is limited to the present fuel which will extend the operations to about the end of this year. A new analysis will be needed for the second fuel loading which is to be supplied by a new fuel supplier. The ACRS has recommended that fuel vendors provide independent overall analyses for reload cores, and it would be expected that the Applicant will submit new evaluations for the continuing operations. Additionally, the ACRS recommended in its October 17, 1973 report on Donald C. Cook Nuclear Plant Unit No. 1 that the NRC Staff independently evaluate the ice condenser behavior under postulated accident conditions. This capability is expected to be achieved by the end of this year. Thus, by the end of the year, two additional efforts for the evaluation of the LOCA-ECCS phenomena will have been completed and the expected thoroughness of these studies in determining the sensitivity of the various parameters should more firmly establish the confidence to be placed in the analyses.
2. The Ice Condenser is a new concept and the Donald C. Cook Nuclear Plant Unit No. 1 is the first plant of its kind in operation. The Applicant has demonstrated competent management in resolving problems that have arisen. Continued operation for the remainder of the first cycle, at a limit less than full power, provides some additional safety margin while the Applicant, through its proposed monitoring of the ice inventory, develops a suitable long-term management program.
3. The ACRS also suggested in its October 17, 1973 report that consideration be given to the possibility of improvements in ECCS effectiveness to provide additional margin. The Applicant effectively responded to all other ACRS recommendations, but not to this item. I believe that this request is reasonable and should be carried out.

4. Generic problems identified by the NRC Staff and by the ACRS have received attention for the review of this application and, for justifiable reasons, action and resolutions to be applied can be deferred; however, in my opinion, an application that seeks to go from a limited power to full power should contribute in some measurable way to the completion of some of these generic problems that are dependent upon analyses to be submitted by the Applicant or to definitions of programs that must be completed for the resolutions.
5. As noted in the ACRS letter, the Applicant's experiences with the APDMS for steady state operation and for several transient power conditions have been favorable. The proposed maximum nuclear peaking factor at full power would be the lowest value for a pressurized water reactor. In this respect, too, Donald. C. Cook Nuclear Plant Unit No. 1 becomes the lead plant in the United States to formalize such control and monitoring procedures. In my judgment, it would be prudent to permit a relaxation of the 81% power limit and to acquire more experience for the remainder of this first core life, but at some power limit intermediate between 81% and 100%.

Additional Comments by Mr. Myer Bender

Normal design practice for any process design is to use bounding values for sizing equipment. It is, therefore, conventional as required by Appendix K of 10 CFR 50 to base ECCS design on the system requirements for the worst LOCA combined with extreme core performance conditions, including the limiting peaking factors as established by nuclear reactivity control and fuel element performance variables. However, the likelihood of a LOCA involving the bounding conditions related to the worst pipe break during the period when the fuel and reactivity control conditions are at their design limits is probably orders of magnitude less than the likelihood of a LOCA which requires a reliable actuation of the emergency core cooling system and is undoubtedly well below the probability of 10^{-6} events per reactor year that would exceed the limits of 10 CFR 100. Consequently, showing the ability to meet all the requirements of Appendix K of 10 CFR 50, while a useful analytical exercise, should not be a governing consideration in determining whether the few early design versions of nuclear power stations that were granted a construction license prior to the promulgation of Appendix K of 10 CFR 50 should be operated at their rated power.

The interests of public safety are better served if those responsible for the operation of the pre-Appendix K of 10 CFR 50 designed power plants demonstrate an attentive interest in assuring that early vintage plants are not operated in a mode that encroaches on the limiting operating conditions that challenge the ECCS capacity for any significant portion of the plant's operating life. The base load operating plans for Donald C. Cook Nuclear Plant Unit No. 1 at rated power are thus consistent with the public safety interest for this pre-Appendix K of 10 CFR 50 vintage design. The Donald C. Cook Nuclear Plant Unit No. 1 should be permitted to operate in this mode at rated power so that the using public can have maximum value from this energy resource.

The emphasis on ECCS improvements for pre-Appendix K of 10 CFR 50 design, such as Donald C. Cook Nuclear Plant Unit No. 1, as well as those of current vintage, should be directed to assuring the adequacy of the emergency core cooling system hardware by verifying its reliability in terms of timely response to smaller LOCA's that will probably occur, assuring continuity of operation during ECCS operating modes, and where appropriate adding diverse and redundant features that will enhance the ECCS reliability under these less extreme but more likely events.

Fundamental understanding of reactor core performance during a LOCA is a matter warranting continuing study through confirmatory research. The thermal-hydraulics evaluation of ECCS performance, while founded on the basic principles of fluid mechanics, involves, as with other fluid mechanics complexities, mainly empirical correlation of heat transfer and fluid

flow parameters that can be verified experimentally only by "cut-and-try" test programs. The experimental and analytical work required to assure an adequate, though not necessarily perfect, understanding of emergency core cooling performance under the full spectrum of accident conditions is therefore an important requirement to be separately satisfied for the purpose of minimizing the risk to public safety arising from erroneous interpretation of the performance analysis. This should be pursued vigorously by the nuclear power industry and in so doing it should address principal attention to the behavior of the emergency core cooling system under less extreme, but more likely, LOCA conditions. This last mentioned evaluation work is of more immediate interest than that associated with the bounding conditions used for design purposes.

References:

1. Donald C. Cook Nuclear Plant Unit No. 1 Startup Test Report, dated October 15, 1975
2. Supplement No. 5 to the Safety Evaluation Report on Donald C. Cook Nuclear Plant Unit No. 1, dated January 9, 1976
3. "Long Term Evaluation of The Ice Condenser System - Results of the December 1974 Initial Ice Weighing Program," by J. G. Feinstein, dated March 1975 (Revised December 7, 1975)
4. "Long Term Evaluation of The Ice Condenser System - Results of the March 1975 Ice Weighing Program," by J. G. Feinstein, dated June 1975
5. "Long Term Evaluation of The Ice Condenser System - Ice Loss Calculations Resulting from the July 1975 and October 1975 Ice Weighing Program," by J. G. Feinstein, dated November 1975 (Revised December 9, 1975)
6. "Axial Power Distribution Monitoring System Experience and Peaking Factor Determination at the D. C. Cook Nuclear Plant Unit No. 1," dated October 15, 1975
7. "NRC Staff Evaluation of Westinghouse ECCS Evaluation Model Changes Documented in WCAP-8622 (TAR-3065), dated January 9, 1976
8. "Westinghouse Evaluation Model, October 1975 Version," WCAP-8622, dated November 1975
9. Donald C. Cook Nuclear Plant Unit No. 1, Technical Specifications, dated January 15, 1976 (Proposed Revision)
10. TWX, BPI to the ACRS, concerning potential fuel failure due to pellet-clad interaction, dated February 4, 1976