02/01/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
HOUSTON LIGHTING & POWER COMPANY	)	Docket No. 50-46[illegible]
	)	
(Allens Creek Nuclear Generating Station, Unit 1)	)	

NRC STAFF RESPONSE TO
JOHN F. DOHERTY'S NINTH SET OF INTERROGATORIES

The NRC Staff responds as follows to the ninth set of interrogatories propounded by John F. Doherty in this proceeding.

9-1. Will a specific "threshold temperature" for the suppression pool water above which unstable steam condensation occurs during an ATWS be calculated for the ACNGS?

a. If so, what options are, in the opinion of the Staff, open to Applicant if that temperature is unacceptably low?

b. Is derating one of the options?

c. Are the vibrations thought to be severe enough to require additional space between fuel rods?

d. Will there be study of the effects of poolswell during the unstable conditions to determine if such swelling may harm control rod drive mechanism hydraulic unit or the Traversing in Core Probe System (TIPS)?

Response

The quencher device proposed by the Applicant to be used for Allens Creek Nuclear Generating Station was selected as a result of tests performed by Kraftwerk Union AG (KWU) in West Germany. The results of these tests indicate

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that unstable steam condensation did not occur even as the local pool temperature approached the nominal boiling point (212°F). However, to ensure that the "threshold temperature" will not occur in plants using a quencher device, the Staff has established a temperature limit of 200°F for suppression pools during all safety/relief valve operational modes including ATWS.

a. The NRC Staff in a meeting with the Advisory Committee on Reactor Safeguards Subcommittee on Anticipated Transients Without Scram (ATWS) on January 25, 1980 indicated that it proposes to proceed with implementation of ATWS requirements in licensing actions without proposing an ATWS rule to the Commission. It will describe the proposed requirements for consideration by the ACRS in its March 1980 meeting. Upon completion of the formulation of those requirements, including modifications in response to comments by the ACRS following its March 1980 meeting, responses to Interrogatory 9.1 and other interrogatories on ATWS may need to be revised. Therefore, this response is limited to information compiled prior to January 25, 1980.

b. See response to a.

c. See response to a.

d. In the event that poolswell cannot be effectively controlled with quenchers and temperature limits, the effects of poolswell on all equipment necessary to mitigate the ATWS would be considered.

The TIP system is not necessary to mitigate an ATWS. Therefore, the only safety concern would be to ensure that the TIP tubes do not leak primary coolant.

The control rod drives are also, by definition, not necessary to mitigate an ATWS. However, damage or further damage to the control rod drives and their supporting components and lines should be minimized, since an ATWS event could otherwise be made worse. Thus, in the hypothetical event of not being able to preclude a significant poolswell, we would review the functionability of the control rod drives, especially the functionability of the alternate rod insertion (ARI) system, as well as the integrity of the pressure boundary.

9-2. Please give the name and a citation to a publicly available document of the foreign reactor where a prolonged blowdown led to loss of integrity of the suppression pool by unstable steam condensation.

Response

The event which resulted in a prolonged blowdown through safety/relief valve and led to some damages on the suppression pool structure was reported in a paper by O. Voigt of KWU.

The paper, "Consequences Drawn From a Stuck Relief Valve Incident at the Wurgassen Power Plant," was presented at the 2nd International Conference on Structural Mechanics in Reactor Technology, September, 1973 in Berlin.

9-3. Has any Staff thought been seriously given to qualifying the HPCI pump to a radiation and steam environment such that in the event of pressure suppression pool loss core melting could still be averted?

a. If so, has Staff:

1. Asked G.E. to study and make a report on such modification.

2. Taken a position that the HPCI pump must be qualified to tolerate the steam and radiation that would enter the auxiliary building in the event of loss of Suppression Pool integrity.

Response

The Staff is not considering qualifying the HPCI pump for the environment following an ATWS with loss of the suppression pool. Loss of the suppression pool could involve qualification of much more than the HPCI pump. Instead, we do not consider suppression pool failure to be an acceptable consequence, and intend to impose sufficiently strict requirements that suppression pool failure need not be considered as a design basis event.

9-4. Has any thought by Staff been given to enlarging or in other ways altering the Condensate Storage Tank (CST) to avoid core melt occurs due to Suppression pool failure?

Response

As stated under Response 9-3 above, the Staff's primary effort is directed toward assuring the availability of the suppression pool.

For LOCA purposes, enlarging the condensate storage tank (CST) was an option which was rejected in favor of another approach when the ECCS were designed. This other approach was to install a "standby coolant supply system."

9-5. Are temperatures above 200°F being considered as possibly acceptable in the event of ATWS with a G.E. BWR-III plant?

Response

Based on the date available to the Staff, a pool temperature higher than the current limit of 200°F has not been considered as acceptable in the event of ATWS. However, we will continue our evaluation as additional data becomes available.

9-6. What other measures are planned to prevent utilities including Applicant from start ups when there is high xenon condition and low or no moderator void? These conditions are though to cause unnecessary short period SCRAMs.

Response

There are no plans at present to require measures beyond those of I&E Bulletin 79-12. The following comments are relevant:

1. The occurrence of a short period scram is, per se, not a safety concern. It is, however, important that the number of these be kept to a minimum in order to reduce the challenges to the protection system.

2. The consequences of continuous withdrawal of a high worth rod during start-up have been analyzed and have been shown to be acceptable (La Salle County Station, Units 1 and 2, FSAR, Section 15.4.1.2 Amendment 36, July 1978, Docket No. 50-373/374).

3. The use of the Banked Position Withdrawal Sequence on Allens Creek will reduce the chances of having high notch worth rod when compared to reactors not using this sequence (where the high notch worths have occurred.)

4. General Electric has issued special recommendations to licensees for operating reactors designed to reduce the chances of occurrence of the high notch worth still further.

In view of the above, it is not deemed necessary at this construction permit stage of review to invoke further measures to prevent short period scrams.

9-7. Using only PWR rods, NUREG/CR-0582, "Evaluating Strength and Ductility of Irradiated Zircaloy, Task 5," states on Page 20,

> One test of the additional Lot 2 material again exhibited a larger hoop strain (60%) at a point other than the burst point; a definite reason for this observed effect has not been determined.

Does Staff oppose this position, to wit, the burst of a fuel rod when subjected to a transient heating burst shows no relationship to locations on the rod where hoop strain is demonstrable?

Response

The referenced testing of Lot 2 Zircaloy cladding (NUREG/CR-0582) involved a total of 16 specimens. Only one of these specimens exhibited a larger hoop strain (60%) at a point other than the burst point (26% strain). This is an infrequent test result. In this case, the anomalous behavior was tentatively attributed to a flaw in the cladding. Since the cladding burst at the thermocouple position, the presence of the thermocouple or its attachment could have affected the final results.

Cladding behavior during burst testing depends on various parameters, mainly heating rate, pressure, and temperature. For the single test discussed above, the NRC burst strain curves (Draft Report NUREG-0630) would have predicted a similar strain (60%) for the test conditions.

9-8. Does the NRC take the position that the most severe loading on the containment steel shell is the LOCA?

a. Has the Commission studied the effects of ATWS on the steel containment shell, such as one proposed by Applicant?

Response

See Response to 9-1.

9-9. Does Applicant's plan for its containment shell have unique or new features not covered in the Standard Review Plan's "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants?"

a. If so, what aspects of Applicant's plans are (or will be) reviewed by Staff?

Response

None were identified in the Staff's review as reported in the Safety Evaluation Report and Supplements to that report. The Staff is currently reviewing changes in the containment that were described in Amendment No. 54 to the Preliminary Safety Analysis Report. Any features found acceptable by criteria other than acceptance criteria of the Standard Review Plan will be identified.

9-10. Have any current standard methods for determining the buckling loads of steel containment vessels that are subjected to unsymmetrical dynamic pressure loads been verified by testing or accurate analysis (See Pg. 1-2, NUREG/CR-0793, where this is largely taken from.)

Response

There is no verification either by testing or by accurate analysis of the current standard methods for determining the buckling loads of steel containment vessels that are subjected to unsymmetrical dynamic pressure loads. Because of this and other concerns raised in NUREG/CR-0743, the Staff initiated a request for proposal and expects to award a contract for a research program to develop a basis for an up-to-date Staff position on buckling of steel containment shells for Staff review of Applicant's designs. The activities of the program will consist of the following:

(a) Perform detailed design analysis of two typical steel containments (one for Mark III, and one for PWR ice condenser).

(b) Develop benchmark problems and their verified solutions to validate computer programs presently used in buckling analysis.

(c) Assess the assumptions and methodology presently used, conduct parametric studies and establish general acceptance criteria for static and dynamic stability of steel containments.

(d) Provide the Staff with all necessary buckling computer programs so that the Staff will be able to independently verify the buckling design of any steel containment.

The various concerns as mentioned in the contention will be addressed by the Staff during the operating license stage of review utilizing the Staff position to be developed. For the construction permit stage of review the Staff's position is as stated in Section 3.8.1 of Supplement No. 2 to the Safety Evaluation Report.

9-11. NUREG/CR-0793 states on page 4-2 "The danger inherent in relying on the results of nonlinear theory is, then, the possibility of using critical loads based on an unconservative distribution of imperfections."

a. At the present are any of the loads computed to predict effects on Applicant's steel shell containment based on non-linear theory?

b. If the answer to a. is "yes," please give the name of the load.

c. Are any steps being taken to assure such computations are conservative, and if so, what are the steps?

d. Is any research on reaching an accurate analysis on these structures expected to be completed before Applicant begins construction of the steel containment shell?

Response

See the response to 9-10.

9-12. In the analysis of axial stress, is the stress presumed to not vary greatly over a distance about equal to a buckle half-wave length? If so, why?

Response

The Staff has not completed its review of Amendment 54 to the PSAR in which changes were made in the methods of analysis for buckling. The results of the review will be reported in a supplement to the SER.

9-13. Does the NRC plan to study and verify the buckling capability of actual containment designs using typical design loading conditions to determine the degree of conservatism of current design methods as is suggested in the conclusion (Pg. 5-2) of NUREG/CR-0793?

Response

See the response to 9-10.

9-14. On P. 7 of NEDO-20,626-1, "Studies of BWR designs for mitigation of ATWS," it states:

> The probability of Relief valves not opening at the setpoint is not addressed because G.E. does not believe the consideration of this failure in addition to the transient initiating failure and the postulated failure to SCRAM is appropriate for the ATWS event."

a. Please cite any directive, letter or other document where Staff has urged consideration of relief valves not opening at setpoint under ATWS conditions.

b. If there are none, please indicate why (if possible) this has never been done.

Response

Our requirements on equipment availability are presented in NUREG-0460, Vol. 3, App. C. If the Applicant demonstrates that relief valve operation meets these criteria, we cannot deny credit for relief valve operation.

However, information equivalent to the effect of relief valve failure will be available because the sensitivity of the ATWS peak pressure calculations to total relief valve capacity was requested by the Staff. (See NRC's (Mattson) 02/15/79 letter to General Electric Company, P. 16). The primary intent of this request was to demonstrate the applicability of the analyses to a range of plants, but the information will also be applicable for this other purpose.

9-15. On page 2 of NEDO-20,626-2, G.E. stated they expected to support the view that plenum (fuel) cladding collapse was ". . . not expected for pressure transients predicted for ATWS events" with experiments scheduled for completion in 1975.

a. Were these experiments completed and their results made public?

b. If answer to a. above is "Yes" please cite the results.

Response

Interrogatory number 9-15 refers to "experiments scheduled for completion in 1975" that were expected to support General Electric's assertion that plenum cladding collapse was ". . . not expected for pressure transients predicted for ATWS events." Our perusal of the statement on page 2 of NEDO-28626-2 indicates that it has been misinterpreted in the sense that analyses, not experiments, were "scheduled for completion in 1975." The results of those analyses are referred to on page 3.6-6 of the attachment to a letter from E. A. Hughes (G.E.)to D. F. Ross (NRC), dated September 27, 1976. In brief, G.E. indicated that, based on a standard G.E. finite element model employed in conjunction with assumptions identified for limiting analyses in G.E. report NEDE-20606, no collapse was predicted.

9-16. Page 6-4 of NEDO 20,626-2 says the peak fuel enthalpy will be 155 cal/gm for the MSIV closure ATWS, and concludes "Thus no cladding perforation is expected during an ATWS as a result of excessive fuel enthalpy."

a. Consulting Exhibit A (infra.) does Staff agree with this statement for:

1. Fuel of more than 20,000 MWd/t burn-up?

2. Fuel of more than 30,000 MWd/t burn-up?

Response

In Table 6-4 of NEDO-20626, the peak enthalpy for the MSIV closure ATWS in a BWR/6 (e.g., Allens Creek) is listed as 150 cal/g, not the 155 value stated in Interrogatory 9-16. More recent General Electric calculations (1976) for this same event indicate that the peak enthalpy is 129 cal/g. The current fuel failure licensing criterion for a BWR reactivity-initiated accident (RIA) is 170 cal/g, and thus the G.E. analyses show that Allens Creek meets the licensing criterion (no failures expected).

As suggested in Interrogatory 9-16 and illustrated in its Exhibit A, there is some current concern that the RIA fuel failure threshold may diminish with increasing irradiation exposure. This effect is being studied and NRC is considering revisions to the licensing enthalpy limits. While revision of this licensing criterion is desirable, it is not essential that it be completed prior to the licensing of Allens Creek for several reasons: (a) RIA physics analyses are, in general, extremely conservative, (b) the MSIV closure ATWS is of low probability, (c) extensive measures are provided for event mitigation

in terms of other ways to limit the radiological releases and (d) the revised peak enthalpy is lower than the licensing values that are under consideration.

9-17. In NUREG-0460, v. 3 on page 29, "Alternative 4. Modifications to Provide Mitigation of ATWS events" is described. Will ACNGS have these modifications and be an "Alternative 4" Plant? (D-8)

Response

The position in Section 15.2 of Supplement No. 2 anticipated that following promulgation of a rule, Allens Creek would be an "Alternative 4" plant. Also, see response to 9-1.

9-18. In NUREG-0460 V. 2, page XI-6, states ". . . BWRs currently appear to have the greatest problem" (with Minimum critical power ratio (MCPR) below a judged safety standard) "with predicted MCPR values as low as 0.82 and 17% of the rods in boiling transition." What steps are being taken to cure this "problem?

a. Are any steps involved with alteration of the fuel rod design?

b. Are any steps involved with changes in the fuel rod assemblies?

c. Describe any other steps being taken that involve any changes in the reactor core.

d. Why are BWR/5's estimated to have but 10.5% failure as compared to 17% for BWR/6 under these ATWS conditions?

Response

The "problem" referred to in Interrogatory 9-18 concerns the fact that, under current NRC fuel design acceptance criteria, fuel rods that are calculated to be in boiling transition are defined as "failed." This is a traditional and usually conservative assumption which has been useful in accommodating failure

mechanisms associated with overheating (e.g., oxidation and embrittlement, overstraining or stress-rupture). We are aware of no steps being taken "to cure the problem"--indeed, having a given number of rods predicted to enter boiling transition during a BWR ATWS event was not unexpected, and as is indicated in NUREG-0460, we do not believe that those rods will necessarily fail. In any cas , it is the intent of NRC ATWS mitigation requirements to preclude violation of 10 CFR 100 radiological dose guidelines regardless of the number of failed rods.

The fact that BWR/5's were estimated to have 10.5% rods in boiling transition as compared to 17% for BWR/6's is a consequence of the different operating characteristics assumed by G.E. in their ATWS analyses. Because BWR/6's have a flatter flux distribution than BWR/5's, more BWR/6 rods (i.e., a larger fraction of the core) will enter boiling transition during an event, such as an MSIV-closure ATWS, that involves a large decrease in the critical power ratio.

9-19. Has G.E. or Applicant developed an ATWS model that contains terms to simulate a pressure pulse upon the closure of the control valve in the steam line. The pressure wave on reaching the core causes a neutron spike.

Response

The Staff has requested General Electric verify the REDY results using the ODYN code which simulates the pressure pulse in the steam line. Gene

Electric has also been requested to make necessary modifications in the ODYN code if needed. Recently the Staff received some information[1/] on the comparison of the results obtained using these two codes. This matter is currently under review by the Staff.

9-20. Has there been any change in Staff concerns regarding stability of the reactor core in response to an event where the recirculation pumps are tripped? The concern is based on the high clad surface heat flux to flow ratios encountered in the transient resulting from the recirculation pump trip.

Response

See Response to 9-1.

9-21. Has General Electric completed its ODYN code? If so, does it account for the significant pressure pulse generated in the steam line for some ATWS?

Response

The development of the ODYN code has been essentially completed. However, some minor modifications may be needed to calculate the entire scenario during an ATWS event. General Electric has been requested to make necessary modifications. It is not expected that the modeling of the steam line dynamics will be changed.

The ODYN code is capable of predicting pressure pulses generated in the steam line and the results had been favorably compared to the Peach Bottom test data.

1/ NEDE-24222, Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3) December 1979.

9-22. In NUREG-0460 V. 2, page XVI-69 it says General Electric has resisted the idea that the number of fuel rods in boiling transition be equated with the number of failed fuel rods, and that therefore MCPR should not be used to determine fuel rod failure.

a. Does G.E. continue this path of thought?

b. Is there still disagreement between NRC and G.E. with regard to the likelihood of pellet/cladding interaction (PCI) failure during the low critical power ratios during an ATWS?

c. Does G.E. still maintain that even though large numbers of fuel rods will enter boiling transition during the worst case ATWS, none would actually fail?

Response

We agree with G.E. that equating the number of fuel rods in boiling transition to the number of failed rods generally results in a conservative estimate of failure. But while G.E. still contends that none of the rods in boiling transition would actually fail, we believe that some could fail by a mechanism (Pellet/Cladding Interaction) not included in their mechanistic analysis.

9-23. Will the ACNGS have redundant SCRAM air header exhaust valves?

Response

See Response to 9-1.

9-24. In NUREG-0460, vol. #2 on page XVI-76, it states:

> If the containment is assumed not to be isolated until ten minutes after the start of the event (Note: that is an ATWS event), and using a realistic site atmospheric dispersion factor of 10^{-4} sec/m^3, we find the offsite dose to be substantially above the guidelines of 10 CFR 100.

a. Is the assumption of 10 minutes for isolation still required for this calculation?

b. Has Applicant or G.E. proposed radwaste system alterations to lessen the danger to this Intervenor of radioactive contamination from the ATWS events?

c. Have their been any changes in these calculations that would alter this finding since April, 1978?

Response

a. While the 10 minute containment isolation time assumed in NUREG-0460, Vol. 3 may appear to be a requirement, this is not the case. The requirement is that the potential radiological off-site consequences of an ATWS event be less than the guideline values given in 10 CFR Part 100.

As you already understand, the time required to isolate the containment during an ATWS event is a very important parameter affecting the potential off-site radiological consequences. Since the final ATWS solution for Alternative 4 plants is still unresolved, the Staff offered the Applicants two approaches to show that the plant as designed would meet the stated post-accident dose guidelines.

The first approach was to show that the containment would isolate much earlier than the 10 minutes assumed in the Staff analysis which you referenced. Credit for such isolation would be given if the Applicant could show that the containment isolation would be automatic and that the necessary isolation signals would be diverse and independent of the plant's

normal reactor protection system isolation signals. Given such credit, the Applicant would still be required to show that, with the automatic isolation, the estimated radiological consequences would be less than the guideline values of 10 CFR Part 100.

The second approach was to show that the estimated radiological consequences would not exceed the guideline values even if the containment were not isolated for long periods of time. The use of 10 minutes came about because Applicants attempted to get credit for operator action to isolate the containment. The Staff has always taken the position that following an ATWS event, credit for operator action could not be given for at least 10 minutes. Further credit for operator action after 10 minutes would be determined by the type and complexity of action required.

Thus, the reason that 10 minutes may have appeared to be a requirement has been that most applicants have chosen the second approach and have assumed that one of the operator's first actions will be to isolate the containment. Having made this assumption, the Applicant must show that the estimated radiological consequences from this approach will not exceed the guideline values of 10 CFR Part 100. If the estimated consequences from this approach exceed the stated guidelines, the Staff will require more rapid isolation of the containment to reduce the offsite consequences below the guideline values.

b. Based upon information to date, there has not been any indication of any planned use or alteration of a plant's radwaste system for the reduction of potential consequences from an ATWS event.

c. The calculation and findings contained in NUREG-0460, Volume 2 have not been reevaluated since they were published in April 1978. Instead, the Staff issued a February 15, 1979 letter to all NSSS vendors which identifed what consequence analyses were to be performed for Alternative 4 plants as well as specifying certain values which should be assumed in the required dose calculations.

9-25. In NUREG-0460, vol. #3 on page 48, it states:

> In a BWR, the Boron injection systems as currently designed are manually actuated and have a small capacity.

a. At present is Applicant's planned standby liquid control system (SLCS) one that meets requirements set before this criticism of April 1978?

b. If not, what changes has Applicant agreed to with its SLCS system?

Response

The present SLCS was designed to meet General Design Criterion 26 (10 CFR 50 App. A). It was not originally designed to be an ATWS mitigating system. Thus, the present system is a system of small delivery rate and is manually operated.

At present, the Applicant has not agreed to upgrade the system. See the Response to 9-1.

9-26. Does Staff concur that the most serious problem with a reactivity insertion accident (RIA) in a BWR is start-up in hot stand-by condition?

a. Is there any study in progress of the effect of forbidding start-up when the reactor is in hot-standby condition?

b. If so, is it generally aimed at determining the highest temperature at which start-up can be done without taking on additional serious consequences in the event of an RIA?

c. If the answer to b. is "yes" are any considerations being made for altering the fuel design, and what alterations? (Note: 9-26 refers to D-3; see: CONF-790441-3, "Light water reactor fuel response during RIA expts.")

Response

The calculated consequences of an RIA (rod drop accident) as reported in safety analysis reports is primarily dependent on the amount of reactivity inserted (worth of the dropped rod). This results from the fact that an adiabatic calculation is performed, i.e., no credit is taken for heat transfer to the moderator during the excursion. Thus the amount of energy calculated to be deposited in the fuel rod as a result of the accident is independent of the thermal-hydraulic state of the reactor. Further, the shape of the scram reactivity curve is based on the assumption that the rods that scram are completely withdrawn from the core. This is the most conservative configuration.

The effect of varying the input parameters for the rod drop accident analysis has been investigated by our consultant, Brookhaven National Laboratories. In particular, the effect of including the moderator feedback (effect of heat transfer to the moderator during the transient) has been studied. For a rod worth of 0.012 change in reactivity the peak enthalpy is approximately 135 calories per gram in the absence of moderator feedback. If the moderator is assumed to be saturated when the accident is initiated (typical of hot standby conditions) and feedback is included then the resultant peak enthalpy is reduced to 35 calories per gram. If 20 degrees Fahrenheit of subcooling is assumed then the resultant peak enthalpy is about 100 calories per gram.

Thus, though it is possible that, for a given peak enthalpy in the fuel, the damage may be greater if the accident is initiated from hot standby conditions than from cold conditions, the reduced enthalpy rise at hot standby conditions compensates for this effect and the safety analysis report analysis is conservative.

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HOUSTON LIGHTING & POWER COMPANY	)	Docket No. 50-466
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(Allens Creek Nuclear Generating Station, Unit 1)	)	

AFFIDAVIT OF CALVIN W. MOON

I hereby depose and say under oath that the foregoing NRC Staff responses to interrogatories propounded by John F. Doherty were prepared by me or under my supervision. I certify that the answers given are true and correct to the best of my knowledge, information and belief.

Calvin W. Moon
Calvin W. Moon
Licensing Project Manager

Subscribed and sworn to before me
this 1ST day of FEBRUARY, 1980.

Nadine C. Lides
Notary Public

My Commission expires: July 1, 1982.

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CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO JOHN F. DOHERTY'S NINTH SET OF INTERROGATORIES" and "AFFIDAVIT OF CALVIN W. MOON" 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, through deposit in the Nuclear Regulatory Commission's internal mail system, this 1st day of February, 1980:

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