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

July 8, 1982

William J. Dircks
Executive Director for Operations

Attn: T. Rehm

Subj: 266TH ACRS MEETING ACTIONS, RECOMMENDATIONS, AND REQUESTS

Based on discussions regarding methods for improved implementation and follow-up of ACRS recommendations, the Committee agreed that a summary of Actions, Agreements, Assignments, and Requests made during each full Committee meeting will be sent to the NRC Staff following each meeting.

Attached in response to this agreement is a list of the requests made at the 266th ACRS Meeting, June 3-5, 1982. This list has the concurrence of the ACRS Chairman and designated ACRS members as will all future items provided for follow up purposes.

Those items in the list "Actions, Agreements, Assignments, and Requests" dated June 24, 1982, that do not deal with requests made of the NRC Staff or that are not pertinent to NRC Staff activities have not been included in this follow-up list.

R. F. Fraley
Executive Director

cc: C. Michelson, AEOD
H. Denton, NRR
R. B. Minogue, RES
R. DeYoung, I&E
J. G. Davis, NMSS
E. Case, NRR
ACRS Members

attachments:
As stated

ACTIONS, RECOMMENDATIONS, AND REQUESTS
266TH ACRS MEETING, JUNE 3-5, 1982

ACRS Interim Report on Midland Plant, Units 1 and 2

NRR 1. The Committee prepared a report to the Commissioners of its review of the Midland Plant Units 1 and 2 regarding the request for an operating license. The Committee concluded that, if due regard is given to comments in the body of the report, and subject to satisfactory completion of construction and staffing, operation at power levels up to 5 percent of full power is acceptable. ACRS recommendation regarding operation at full power has been deferred until the Committee has had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question of natural circulation in the presence of a small break LOCA. Specific recommendations/comments in the body of the report addressed the following follow-up items:

- Limit plant operation to 5% of full power following satisfactory completion of construction and staffing and due regard to comments in the Committee's report.

- Confirmation of the overall quality assurance program and the seismic design basis, including the adequacy of protection against liquifaction of the granular portions of the soil fill in the event of strong vibratory motions accompanying an earthquake.

- A broader assessment should be arranged by the NRC Staff of Midlands design adequacy and construction quality with emphasis on installed electrical, control, and mechanical equipment as well as piping and foundations. A report should be provided to the ACRS regarding design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

- All systems and components important to decay heat removal should be carefully evaluated for their ability to accomplish necessary functions in the unlikely event of lower-probability earthquakes more severe than the proposed SSE, in order to provide the necessary degree of assurance.

- Resolution of a high point vent on the reactor vessel head.

- Further applicant review of the potential for providing indications of water content or level within the reactor vessel.

- The NRC Staff should institute an augmented audit of plant operations, at least during the early years of operation at power.

- The ACRS desires the opportunity to review the plant specific probabilistic risk assessment for Midland with assistance from the NRC Staff.

- Availability of natural circulation in the presence of an interrupted or continuing small break loss-of-coolant accident in connection with B&W plants.

- The ACRS wishes to be informed regarding the results of the plant-specific systems interactions study.

- Additional prudence is appropriate for the Midland Plant regarding resolution of the ATWS issue and other Unresolved Safety Issues.

- There should be active participation of Midland Plant personnel in emergency response procedures developed on the basis of an assumed failure at the Dow Chemical Plant.

- Safety implications of control systems awaits completion of NRC Staff Task Action Plan A-47.

- Complete resolution of instrumentation to follow the course of an accident.

- Generic resolution of environmental qualification of equipment as it applies to Midland.

- Missile issue, the ACRS wishes to be kept informed.

[Note: W. Kerr did not participate in the review of the Midland Plant.]

ACRS Report on Pressurized Thermal Shock

NRR 2. The Committee prepared a report to the Commissioners of its review of the current status of the pressurized thermal shock problem. The ACRS noted lack of sufficient information to evaluate the adequacy of an approach by the NRC Staff to develop a regulation based upon a combination of deterministic and probabilistic analyses. Specific recommended actions in the short term for plants identified to have potential pressurized thermal shock problems are as follows:

   - Make certain that the metallurgical properties of the vessel beltlines are established adequately with respect to fracture toughness.

   - Determine the most effective in-service inspection capability for the beltline using current technology and apply it at the next refueling shutdown, if practical, if not accomplished already.

   - Provide effective operator training to avoid thermal shock and provide capability to diagnose events that could cause thermal shock.

- Examine the depressurization capability for the existing plants and train operator when and how to use it.

- Provide a demonstration of pressure vessel annealing to recover fracture toughness.

ACRS Comments on Proposed Policy Statement on Safety Goals for Nuclear Power Plants (NUREG-0880, "A discussion Paper")

3. The Committee prepared a report to the Commissioners of its review of NUREG-0880, A Discussion Paper, recommending that final action on adoption of a policy statement on safety goals should be contingent upon proper evaluation and agreement on the implementation plan. The ACRS plans to provide further comments to the Commission after reviewing the Staff plan for implementation. M. Bender and H. W. Lewis appended additional comments. In the body of the ACRS report will also be found responses to the four questions raised by the Commission.

NRR

Response to Commissioner Gilinsky Regarding Seismic Design Suggestions by Professor Paul Jennings

4. The Committee prepared a report to Commissioner Gilinsky recommending that the suggestions by Professor Paul Jennings on seismic design be considered within the context of a broad review of the NRC Staff's current seismic design practices including the NRC Staff's reassessment of Appendix A to 10 CFR 100. The ACRS suggested that Professor Jennings be invited to participate in this review.

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Consideration of Seismic Events in Emergency Planning

5. D. W. Moeller indicated that the NRC Staff is developing a position paper with regard to consideration of seismic events in emergency planning at nuclear power plants. He suggested that the Committee wait for issuance of the position paper and review the draft at that time.

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Backfit of Feedwater Overfill Protection to Operating Reactors

6. R. Mattson, NRC Staff, indicated that steam generators should be protected from overfill by main or auxiliary feedwater flow. He added that equipment to provide this protection of overfilling should be safety grade if flooding of the steam lines is an unanalyzed event. D. Okrent noted that this issue is particularly important on B&W plants, including Midland, because of their control sensitivity and questioned the lack of urgency at issuing a backfit requirement for operating plants. D. Okrent requested that the Staff provide a written answer within the next month regarding its position

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ACTIONS, RECOMMENDATIONS, AND REQUESTS
266TH ACRS MEETING, JUNE 3-5, 1982

with respect to the issue of feedwater overfill protection. J. Ebersole requested that the NRC Staff include in its answer a statement regarding the consequences of continuing to pump cold main feedwater into the plant in the event of a main steam line failure, whether this procedure can lead to a severe secondary transient leading to the pressurized thermal shock problem in the reactor vessel.

Turbine Missile Problem

7. As part of the Midland review, J. Ebersole questioned the position of the NRC Staff regarding the problem of sticking of the turbine stop valve and control valve failure which could lead to disc failures due to turbine overspeed. P. G. Shewmon also asked the NRC Staff to explain their general approach with regard to the turbine missile strike probability P_2 and the damage probability P_3.

NRR

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

June 8, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS INTERIM REPORT ON MIDLAND PLANT, UNITS 1 AND 2

During its 266th meeting, June 3-5, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Consumers Power Company for a license to operate the Midland Plant, Units 1 and 2. This application was also considered at Subcommittee meetings held on April 29, 1982 in Washington, D. C., on May 20-21, 1982 in Midland, Michigan and on June 2, 1982 in Washington, D. C. On May 20, 1982 members of the Subcommittee toured the plant. In the course of these meetings the Committee had the benefit of discussions with representatives and consultants of Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, the Nuclear Regulatory Commission Staff, and members of the public. The Committee also had the benefit of the documents listed below.

The ACRS reported on June 18, 1970 regarding the construction permit application for the Midland Plant; on September 23, 1970 regarding several amendments to the application; and on November 18, 1976 regarding applicable generic matters.

The Midland Plant site is located on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site. There are about 2000 industrial workers within one mile of the site, and the estimated 1980 population was about 51,400 residents within five miles of the site. This makes the Midland site one of the more densely populated sites at distances close to the Plant.

Each of the two Midland units employs a Babcock and Wilcox designed nuclear steam supply system rated at 2468 MWt with a stretch power rating of 2552 MWt. The Midland Plant is unique in that the heat generated will be used not only to produce electricity but also to produce process steam for the Dow Chemical Company plant via a tertiary system.

The Midland Plant has been the subject of several major problems related to quality assurance during plant construction. One of these problems relates to the soil fill under several safety-related structures. The

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deficiencies relating to soil fill have led to excessive settlement and some cracking of these structures, and have also introduced questions concerning the adequacy of protection against liquefaction of the granular portions of the fill in the event of strong vibratory motion accompanying an earthquake.

The Applicant has proposed and is implementing, under close surveillance by the NRC Staff, remedial measures with regard to the foundation deficiences. We are generally satisfied with the approach being taken, subject to confirmation of the overall quality assurance program and the seismic design basis. Both of these items are discussed below.

With regard to quality control of design and construction, the report of the NRC Staff's Systematic Assessment of Licensee Performance (SALP) review for the period July 1, 1980 to June 30, 1981 revealed deficiences in the installation of piping and piping suspension systems, in the pulling of electrical cables, and in the handling of problems relating to soils and foundation. Deficiencies by the Applicant in the handling of soils-related matters have continued to occur, subsequent to issuance of the SALP report. We believe that the NRC Staff is handling the corrective actions for specifically identified quality assurance deficiencies in an appropriate manner.

In view of the overall concern about Midland quality assurance the NRC should arrange for a broader assessment of Midland's design adequacy and construction quality with emphasis on installed electrical, control, and mechanical equipment as well as piping and foundations. We wish to receive a report which discusses design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

Our reservation concerning seismic design relates to the lack of adequate assurance that the Midland Plant will be capable of accomplishing shutdown heat removal for low probability earthquakes more severe than the safe shutdown earthquake (SSE). The Midland seismic design basis at the construction permit stage corresponded to a MMI VI, peak ground acceleration of 0.12g, employing a modified Housner spectrum. For the operating license review, the NRC Staff has reevaluated the original seismic design basis and the Applicant and the NRC Staff have agreed on the use of site-specific analyses which have led to increases in the design response spectra for frequencies above about 2 cycles/sec.

Historically, no earthquakes stronger than the newly proposed SSE have occurred within 200 miles of the Plant. However, expert opinion differs widely on the exceedance frequency of the proposed SSE and on the severity at the site of earthquakes whose likelihood is less than 1 in 10^4 or 1 in 10^5 per year.

The Applicant is currently reevaluating by selective audit the seismic capability of the plant, as originally designed, to withstand the revised SSE. Measures taken to assure safe shutdown in the event of an earthquake include the use of dewatering to reduce the potential for soil liquefaction. We recommend that all systems and components important to decay heat removal be carefully evaluated for their ability to accomplish necessary functions in the unlikely event of lower-probability, more severe earthquakes in order to provide the necessary degree of assurance. This matter should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed about the resolution of this matter. We believe that any recommendations for changes in the plant resulting from this evaluation should be implemented by the end of the second refueling outage.

The Applicant has agreed to provide core exit thermocouples, a hot-leg-level measurement system, and subcooled margin monitors as instrumentation to detect inadequate core cooling. Consumers Power Company also plans to include a remotely operable vent on top of both inlet loops to the steam generators; however, Consumers has not committed to supply a high point vent on the reactor vessel head. This matter should be resolved in a manner satisfactory to the NRC Staff. The ACRS recommends that the Applicant review further the potential for providing indications of water content or level within the reactor vessel.

The staff of the Applicant includes many personnel who have had nuclear power plant experience. However, operating experience with this B&W type power reactor is limited, and the NRC Staff is requiring that at least one person having experience on a large commercial PWR be included on each shift for one year. We support the NRC Staff position.

The Applicant's experience with the operation of nuclear power plants should, in principle, place Consumers in a favorable position to provide continuing, careful oversight of the operations at the Midland Plant. In view of some prior adverse operating experience at the Palisades Plant however, we recommend that the NRC Staff institute an augmented audit of operations at Midland, at least during the early years of operation at power.

We have reviewed the evaluation made of the tertiary process steam system for use by Dow Chemical Company. This system appears not to impose any unacceptable impacts either on the safe operation of the Midland Plant or on the people working at the Dow Chemical Company.

The Applicant has undertaken an effort to have a probabilistic risk assessment (PRA) performed for the Midland Plant and stated that the results will be available in the fall of 1982. We believe it desirable to have plant-specific PRAs performed for each commercial nuclear power plant and that it is particularly appropriate for the Midland Plant because of its relatively high, close-in population density. We wish to have the opportunity to review the Midland PRA with assistance from the NRC Staff, and to offer comments or recommendations as appropriate. We do not believe that this review need delay licensing of the Midland Plant for operation.

Recently, questions have come to light in connection with B&W plants concerning the availability of natural circulation in the presence of an interrupted or continuing small break loss-of-coolant accident. We wish to see a proposed NRC Staff resolution of this issue.

The Applicant described an extensive systems interactions study being undertaken for the Midland Plant. We wish to be informed of the results of this study.

We believe that, in view of the population density near this plant, additional prudence is appropriate for the Midland Plant in the resolution of the ATWS issue and other Unresolved Safety Issues.

We endorse the participation of Dow Chemical Company plant personnel in emergency procedures developed on the basis of an assumed failure at the Midland Plant. Similarly, there should be active participation by Midland Plant personnel in emergency procedures developed on the basis of an assumed failure at the Dow Chemical plant. The Applicant and the NRC Staff should promote continued coordination of these types of relationships, as well as those involving appropriate state and local groups to assure that the capability for an effective emergency response is developed and maintained.

With regard to the eleven items identified in the ACRS Supplemental Report on Midland Plant, Units 1 and 2 dated November 18, 1976, we have the following comments. The issues related to vibration and loose-parts monitoring, potential for axial xenon oscillations, behavior of core-barrel check valves during normal operation, fuel handling accidents, effects of blowdown forces on core internals, LOCA-related fuel rod failures, and improved quality assurance and in-service inspection for the primary system have all been resolved or are in a confirmatory stage of being resolved. Separation of protection and control equipment has been accomplished in an appropriate manner; however, the safety implications of control systems remains an Unresolved Safety Issue directly applicable to Midland. Resolution awaits completion of the NRC Staff Task Action Plan A-47. The effect of ECCS induced thermal shock on pressure vessel integrity has been resolved in part; however, the Unresolved Safety Issue on pressurized thermal shock will apply. Environmental qualification of equipment remains a generic issue which is under review by the NRC Staff and whose resolution will apply to the Midland Plant. Instrumentation to follow the course of an accident has been resolved in part by the development of revised Regulatory Guide 1.97. We do not believe that licensing of the Midland Plant for operation need await further resolution of any of the eleven issues discussed above.

The various other matters identified by the NRC Staff as open or confirmatory in the Safety Evaluation Report should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept advised concerning resolution of the turbine missile issue.

The ACRS believes that, subject to satisfactory completion of construction and staffing and if due regard is given to the comments above, the Midland Plant, Units 1 and 2 can be operated at power levels up to 5 percent of full power with reasonable assurance that there is no undue risk to the health and safety of the public.

We defer our recommendation regarding operation at full power until we have had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question regarding natural circulation in the presence of a small break LOCA.

Dr. Kerr did not participate in the Committee's review of this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman,

References:
1. Consumers Power Company, "Midland Plant Units 1 and 2 - Final Safety Analysis Report" including Amendments 1-43
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Midland Plant, Units 1 and 2," NUREG-0793, dated May 1982
3. U.S. Nuclear Regulatory Commission, "NRC Licensee Assessments," NUREG-0834, dated August 1981
4. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Response to Draft SALP Report, dated May 17, 1982
5. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Quality Assurance Program Update, dated April 30, 1981

6. Letter from J. Hind, NRC, to J. Cook, Consumers Power Company, Subject: Systematic Assessment of Licensee Performance (SALP), dated April 20, 1982
7. Letter from J. Cook, Consumers Power Company, to H. Denton, NRC, Subject: Summary of Soils-Related Issues at the Midland Nuclear Plant, dated April 19, 1982
8. Letter from K. Drehobl, Consumers Power Company, to D. Fischer, ACRS, Subject: Midland Project Soils Information, dated April 12, 1982
9. Statement of Ms. M. Sinclair to ACRS, dated June 4, 1982
10. Letter from B. Stamiris to Dr. D. Okrent and ACRS Members, Subject: Midland OL Review, dated May 29, 1982
11. Letter from M. Sinclair to Dr. P. Shewmon, ACRS, Subject: Midland OL Review, dated May 28, 1982
12. Statement by Dr. C. Anderson to ACRS Midland Plant Subcommittee dated May 20-21, 1982
13. Statement by Ms. M. Sinclair to ACRS Midland Plant Subcommittee dated May 20-21, 1982
14. Letter from B. Stamiris to D. Fischer and ACRS Members, Subject: Soil Settlement and QA Issues, dated May 20, 1982
15. Letter from M. Sinclair to Dr. C. Siess, ACRS, Subject: Midland Soil Settlement, dated April 26, 1982

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

June 7, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

Subject: ACRS REPORT ON PRESSURIZED THERMAL SHOCK

During its 266th meeting, June 3-5, 1982, the Advisory Committee on Reactor Safeguards completed its review of the current status of the pressurized thermal shock problem (PTS). The NRC Staff is developing a regulation based on a combination of deterministic and probabilistic analyses to establish regulatory requirements concerning pressurized thermal shock. The ACRS has not been provided sufficient information to evaluate the adequacy of this approach.

The ACRS does not believe there is a need for any immediate plant modifications to permit continued operation of the plants which have been identified up to now as having potential PTS problems.

The most beneficial actions for these plants in the short term would be to:

1. Make certain that the metallurgical properties of the vessel beltlines are established adequately with respect to fracture toughness.

2. Determine which is the most effective in-service inspection capability for the beltline that current technology can provide. For those welds of principal concern, inspection should be accomplished, if practical, at the next refueling shutdown using such techniques, if such inspection has not previously been accomplished.

3. Provide effective operator training to avoid pressurized thermal shock and provide capability to diagnose events that could cause it.

4. Examine the depressurization capability for these plants and train operators when and how to use it.

5. Provide a demonstration of pressure vessel annealing to recover fracture toughness.

There are many intricacies associated with evaluation of pressurized thermal shock consequences that deserve attention, but the above actions would be the most effective contributors to assuring that pressurized thermal shock does not create public safety problems.

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The ACRS plans to continue its review of pressurized thermal shock and the related NRC Staff program. The Committee will report further at an appropriate time.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

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

June 8, 1982

Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Commissioner Gilinsky:

This is in response to your request for comments, dated December 3, 1981, regarding a letter from Professor Paul Jennings on proposed changes in seismic design methodology.

We believe that Professor Jennings' suggestions are of interest and merit careful consideration. We recommend that they be considered within the context of a broad review of the NRC Staff's current seismic design practices. The NRC Staff is planning to reactivate their reassessment of Appendix A to 10 CFR Part 100 in the near future. We believe that this would be an appropriate forum for these discussions and recommend that Professor Jennings be invited to participate. We note that the NRC Staff in licensing actions and in their past evaluations of seismic design practices has addressed some of the issues on which Professor Jennings has commented and has in some instances adopted approaches similar to what he has suggested. We expect to be involved in the planned reassessment of Appendix A to 10 CFR Part 100 and will further consider Professor Jennings' suggestions at that time.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

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

June 9, 1982

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: COMMENTS ON PROPOSED POLICY STATEMENT ON SAFETY GOALS FOR NUCLEAR POWER PLANTS (NUREG-0880, A DISCUSSION PAPER)

We commend the Commission and the NRC Staff for their efforts to develop safety goals for nuclear power plants. However, we believe that the safety goals as described in the Proposed Policy Statement represent only a step in the process. We believe that the matter of how the proposed policy statement is to be implemented is so important that the Commission should not take final action on adopting a policy statement on safety goals until there has been ample time for evaluation and agreement on the implementation plan.

We note that the policy statement proposes a set of qualitative safety goals. Although much of the discussion of safety goals has focused on the desirability of trying to develop quantitative goals, we believe the proposed qualitative goals are a useful statement of the position of the Commission on the risk to which it believes it would be acceptable for the public to be exposed by accidents in nuclear power plants. The numerical guidelines provide the public additional explanation of what the Commission means by the qualitative goals.

However, we believe it is important that explicit usable quantitative goals be provided for the industry and the NRC Staff as a guide for meeting the qualitative goals expressed to the public, including guidance on how to deal with issues involving large uncertainties. We are convinced that, for the present, a clear distinction should be made between the two sets of goals, although they should be generally consistent. If Probabilistic Risk Assessment (PRA) were sufficiently developed, the two objectives might be met by the same statement or the same set of goals. We do not believe PRA is sufficiently developed at this time for that purpose. For this reason, quantitative goals for the use of the NRC Staff and the industry may have to be more limited in scope than those suggested by the numerical guidelines in the policy statement. We recommend that the numerical guidelines be design-oriented. For example, numerical specifications on required reliability of core cooling and of containment function may be an appropriate starting point. In some cases guidance will also be needed on the influence of site population differences.

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A principal difficulty facing the development and implementation of a more quantitative approach to nuclear power plant safety lies in the large uncertainties inherent in many aspects of PRA. This fact lends support to those who advocate a go-slow attitude to the adoption of a quantitative approach. On the other hand, decisions have to be made; properly used, PRA may help in improving them. Furthermore, despite the absence of a quantitative NRC safety policy, the NRC staff and the industry are applying PRA methods to decision-making in an increasing number of safety issues with the aid of ad hoc quantitative safety criteria and with a widely varying degree of quality and credibility in the PRA analyses being utilized. We believe that the Commission needs to provide a proper balance between the conflicting opinions on whether or not to go slowly, and to assure that the necessary control and guidance exists for the current and future use of PRA.

In what follows, we make several general comments on this subject and then respond to the four questions raised in the Commission's proposed policy statement.

GENERAL COMMENTS

1. We believe that, rather than exclude sabotage from any consideration of the risk from accidents in nuclear power plants, the NRC should adopt a position resembling the following:

> Partly because of the great difficulty in quantifying the risk arising from sabotage and especially because the measures included to prevent or minimize the chance of a serious accident due to sabotage are best not discussed in public, the NRC will not include an evaluation of the risk from sabotage in any analysis intended to provide a quantitative estimate of the risk from accidents in light water nuclear power plants. However, the NRC intends to assure that the contribution to risk which arises from sabotage is compatible with its overall safety goals and will develop consistent design and operational requirements.

2. The draft statement appears not to distinguish between plants in operation or under construction ("Existing Plants") and plants yet to be designed. We believe that one can and should consider distinguishing between such plants in the requirements applied to both siting and design. For example, probabilistic risk assessment of some existing plants may yield computed core melt frequencies larger than would be deemed acceptable for plants yet to be designed. Such a result should not be unanticipated and an approach for dealing with it should be addressed in the policy statement and the Staff implementation plan.

3. Numerical guidelines on individual risk should address the risk to the individuals subject to the largest risk of exposure. However, for operational convenience it may be acceptable to express such a guideline in terms of the average risk to an assumed biologically average individual, living within one mile from the site boundary, if it can be assumed that there would not be significant variations in risk to such individuals over this region and that the risk would be less at distances greater than a mile.

The use of a quantitative guideline for individual risk of early death of 0.1% of the risk of accidental death from all causes (with a similar guideline for latent cancer) provides a useful way of placing the risk in perspective; however, it may lead to risk limits which are more stringent than necessary if they are to be met with a reasonable degree of assurance.

As proposed in the policy statement, the provisional numerical risk guidelines permit nuclear power plants in more densely populated areas to impose greater total societal risks than those in less populated areas. Furthermore, the proposed quantitative societal guideline provides no incentive for the use of less populated sites, other things being equal.

We recommend that the societal risk guideline as proposed in the policy statement be replaced by one that places a numerical limit on the statistical health effects per 1000 MWe reactor year (or some similar unit). The policy statement should also point out the protection to society that is implicit in the numerical guidelines for risk to the individuals with the greatest risk of exposure.

4. The large uncertainties inherent in PRA are well recognized and are acknowledged in the proposed policy statement. These uncertainties make the use of PRA in decision-making (which occurs already within the NRC) subject to large differences in the results obtained by different groups of analysts for the same accident scenario. These uncertainties also permit abuse of the methodology to obtain a result which supports a predetermined position by selective choice of data and assumptions. We believe that the Commission needs to consider what requirements it should develop with regard to depth and independence of peer review, for both risk analyses conducted by licensees and those performed by or for the NRC Staff.

5. We agree that it is appropriate at this time to consider the risk posed by nuclear power plant accidents separately from that posed by other parts of the fuel cycle. However, because of the interest that exists in the rest of the fuel cycle, and because comparisons with other fuel cycles will be made in the choice of methods for energy generation, we conclude that the statement should either indicate that the risk of the rest of the cycle is small or that it will be addressed later.

6. We have not had the benefit of seeing an NRC Staff plan for implementation of the proposed qualitative safety goals and numerical guidelines. When the Staff plan is available for review, we will provide further comments to the Commission.

Response to Commission Questions

Question 1: Should the benefit side of the tradeoffs include, in addition to the mortality risk reduction benefits, the economic benefit of reducing the risk of economic loss due to plant damage and contamination outside the plant?

Response to Question 1: The proposed benefit-cost guideline of $1000 per man-rem averted out to fifty miles from nuclear power plant accidents places a larger value on averting premature death than is generally used by the Department of Transportation or other federal agencies where this attribute is explicitly discussed. However, genetic effects are not included, nor are psychological effects on health, and these might be considerable. Also, the man-rem incurred at distances greater than fifty miles are likely to be comparable to or greater than the portion within 50 miles. Further studies should be made to provide better quantitative insight into the benefit to be attributed to a reduction in health effects.

Economic loss due to plant damage and to contamination outside the plant would be as real a loss to society as direct health effects. We therefore recommend that the ALARA criterion include benefits from reduction of offsite and onsite economic losses.

Section IV. Implementation, of the proposed policy statement, indicates that the ALARA benefit-cost guideline may be used as one consideration in backfitting plants previously approved for construction or operation. It is not specifically stated that the ALARA criterion would also apply in the design of new plants for the evaluation of possible cost-effective features which could reduce risk to levels significantly below the quantitative guidelines. We believe that the Commission should state that the ALARA criterion is also intended to apply in this way for new plants.

Question 2: Should there be added a numerical guideline on availability of containment function, given a large-scale core melt?

Response to Question 2: Yes, but the approach which should be taken to address this issue is probably different for plants yet to be designed than for plants already approved for construction or operation, i.e., existing plants. Except for some NTCP plants and the proposed Floating Nuclear Plant, the plant containment designs do not explicitly include consideration of measures to cope with or mitigate the consequences of accidents involving large-scale core melt.

For plants already approved for construction or operation, the containment capability could be assessed as part of a plant-specific assessment for each plant. This study would include the merits, costs, and disadvantages of possible improvements in the capability of the existing containment to cope with or limit the consequences of core melt accidents. Following these studies, the NRC would make decisions on a case-by-case basis in the light of the safety policy it eventually adopts, plus other relevant factors.

For plants yet to be designed, it may be practical to set containment performance standards for accidents leading to large-scale core melt, but not automatically involving a direct loss of containment integrity by means, for example, of a large missile. Assuming that the frequency of core melt accidents that are directly coupled with an early loss of containment integrity is and must be kept very low, recent studies indicate that it is practical to establish stringent performance requirements on containment capability for other core melt accidents. We believe that additional study is needed before numerical guidelines are set for the containment performance of future plants.

Question 3a: What further guidance, if any, should be given for decisions under uncertainty?

Response to Question 3a: It is to be expected that large uncertainties will remain a continuing aspect of probabilistic risk assessment.

Without fairly detailed agreement on how calculations are to be made, including how uncertainties are to be incorporated and indicated, numerical guidelines will be lacking in operational significance. Thus, specifications on uncertainty are an essential part of the numerical guidelines.

These guidelines should include consideration both of the design and operational stages. That is, methods need to be specified for calculating the expected behavior at the design stage, and methods must also be developed for determining whether operating plants are meeting the guidelines.

However, there exists a conflict between a desire to provide sufficiently prescriptive rules on how to conduct a PRA so that it can more readily be reviewed and evaluated, and the acknowledged state of immaturity of methods development and lack of adequate data. It may be possible to obtain an apparent conformity of results by means of a fictitious reduction in uncertainty by the use of prescriptive rules; the ACRS does not recommend such an approach.

The Commission should direct the NRC Staff to include an assessment of uncertainties in all PRA results and should provide broad guidance on how to judge and proceed, both when the uncertainties seem fairly well defined and when the most plausible result is that it is very difficult to define meaningful uncertainties.

Question 3b: What further guidance, if any, should be given on resolution of possible conflicts among quantitative aspects of some issues?

Response to Question 3b: The specifications for implementation of the policy statement must include enough detail on the decision-making process that it can be used. If, as is proposed, licensing decisions will continue to be based on existing regulations initially, a start toward the process might be made by emphasizing a few key systems on which it might be possible to reach a consensus with a minimum of controversy. Thus, for example, one might select systems such as:

(1) The Reactor Protection System or subsets thereof,

(2) The Decay Heat Removal System or subsets thereof,

(3) The Electrical Systems, AC (both offsite and emergency onsite) and DC, and

(4) Containment systems.

It will be relevant to embed such analyses within an understanding of the total plant. Experience in dealing with these topics in detail may suggest a more general approach to the overall problem.

We suggest that, on a concurrent basis, the NRC Staff undertake the evaluation of non-binding, global PRAs for specific plants, employing a process which has the benefit of all the independent reviews and decision-making steps one would employ if the process were subject to challenge in the courts.

Question 3c: What approach should be used with respect to accident initiators which are difficult to quantify, such as seismic events, sabotage, multiple human errors, and design errors?

Response to Question 3c: The approach for dealing with risk contributors should depend upon the context of the particular regulatory decision. The Commission intention should be that the goals have been met when all accident initiators have been accounted for. However, some factors influencing risk, such as sabotage and management organization, are not likely to be quantified satisfactorily. Others are difficult to quantify, and large uncertainties are associated with their contribution to risk. The implementation plan for the safety goals should address all of these contributors to risk to ensure that decisions are consistent with the safety goals. Those factors which cannot at present be treated adequately by PRA methods should be treated by other means.

Question 3d: Should there be definition of the numerical guidelines in terms of median, mean, 90 percent confidence, etc.? If so, what should be the terms?

Response to Question 3d: The answer to this question is inseparable from the question of how uncertainties should be incorporated into the analysis and decisions in general. The mean value of a risk distribution appears to be the most meaningful parameter for comparison with the numerical guidelines. However, it is important to provide an accurate representation of all the uncertainties, both random and systematic, which enter into the analysis. Where subjective opinion is a principal factor in the quantification process, the dependence of the results on the particular choice of input parameters or analytical approach should be carefully documented with the help of sensitivity analysis.

If the systematic or random uncertainties are so large or so poorly known as to make it difficult to define a meaningful mean value, that state of affairs should also be documented. When uncertainties are large, so-called best estimate values may not have any real meaning, and heavy reliance should not be placed on a direct comparison of the PRA results to quantitative guidelines or goals.

Question 3e: Should the Staff action plan include further specification of a process which will lend credibility to the use of quantitative guidelines and methodology? If so, what should be the principal bases and elements of such guidance?

Response to Question 3e: A clearly stated safety philosophy and an implementation plan that is balanced and that allows for expected contingencies is important. In view of the anticipated large uncertainties and considerable differences in the results provided by differing groups, applicability and usefulness will depend on the ability to obtain a process that yields reasonable decisions and is perceived by all parties concerned, including the public, to be fair.

Question 3f: On what basis should the numerical guidelines be applied to protection of individuals? Should they be applied to the individual at greatest risk, or should they be used in terms of an average risk limit over a region near the plant? Any comments or suggestions pertaining to the present discussion of this topic (or other specifics) would be welcome.

Response to Question 3f: This is discussed in general comment No. 3.

Question 4: Should there be specific provision for "risk aversion"? If so, what quantitative or other specific provision should be made?

Response to Question 4: The proposed safety goal policy statement not only does not include any element of risk aversion, in its current form it permits a reactor located in a region of relatively higher population density to impose greater societal risks than a reactor at a remote site. Hence, the proposed safety policy provides a kind of disincentive to the use of relatively less populated sites. The proposed safety policy also provides no incentive to consider sites offering less likelihood of seriously affecting a major societal resource, such as an important aquifer or estuary.

We believe that society is risk averse at least to the extent that it prefers not to introduce the potential for very large accidents for activities other than those essential to society. Where such activities are needed, society wants additional assurance that the safety design has been well conceived and executed and that as far as practical the best available technology is used.

We suggest that the NRC adopt a safety policy that explicitly includes measures intended to reduce the likelihood of large accidents. Such a policy might read as follows:

- All practical measures should be taken to reduce the likelihood of an accident that can seriously degrade the reactor core.
- Containment capability should be provided, as far as practical, for a wide spectrum of severe accidents.
- Emergency plans should be carefully prepared and developed.
- Sites for new nuclear power plants should be selected so as not to impose an unnecessary risk to people or important resources.

Additional comments by members M. Bender and H. W. Lewis are presented below.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

Additional Comments by ACRS Member M. Bender

The general approach described in NUREG-0880 is of considerable value as a statement of principles used to guide regulatory policy and implementation actions; however, the numerical values in the policy statement have questionable meaning and are of use mainly as a basis for discussion. The proposed value of 10^{-4} core melts per reactor year is consistent with various probabilistic risk assessment studies that have been performed including the reactor safety study, WASH-1400, and the recent work on the Zion and Indian Point plants. The definition of core melt remains unclear. However, if we interpret the meaning to be an event that can be contained by the types of containments being provided for already licensed plants and those proposed for construction permits up to now, I believe that the public safety risk as measured by the potential for prompt fatalities in accidents of this type and the potential for cancer-causing effects is acceptable with the proviso that the designs do perform as intended for the purpose of controlling and containing the postulated accidents. Events comparable to and with more severe consequences than the TMI-2 event would fit within this definition. The policy should state whether existing plants can meet the proposed goals.

The ALARA principle included in the policy statement is difficult to interpret and needs work on implementation approaches before being given a regulatory trial. If it is intended to show the benefit from reduction of low level radiation effects from serious accidents, its benefits should be measured by realistic values rather than upper bound estimates of radiation effects and probably should relate to background radiation levels and exposure from other radiation sources. Furthermore, specially engineered features to satisfy ALARA principles should be considered in the context of need probability. This approach would be very difficult to apply, and I do not believe that public safety would be enhanced by its inclusion in the safety goal policy. It is not analogous to the 10 CFR 50 Appendix I application which is claimed to provide precedent.

The numerical values concerning cancer-related risk in the policy statement have an unclear meaning. If they apply only to the small number of people tethered within the one-mile exclusion boundary whose risk exposure is expected to be only 0.1% of those risks leading to cancer-caused death then the actual consequential death effects would be unmeasurably low. Thus, the basis would be only useful for abstract discussions. If, on the other hand, the intent is to interpret the value as permitting nuclear plants to contribute 0.1% of cancer-caused deaths to the population within a 50-mile radius, then it would be unacceptably high. The intent of the numerical goal, if used, should be to indicate the degree to which nuclear plants contribute to cancer-related mortality rates and the computational application of the numerical value should be illustrated.

Additional Comments by ACRS Member M. Bender (Cont'd)

The public mortality and human health damage effects actually experienced from commercial nuclear power plants are so low that they are not measurable and cannot be related to human health and mortality statistics. All studies by statistical experts in human health effects indicate that, if there are nuclear plant radiation-caused cancer mortality effects, they are totally masked by other causes of cancer which are directly correlatable with human health statistics. No identifiable catastrophe would uniquely affect mortality rates within the one-mile exclusion boundary and the plant operating staff would in catastrophies represent a more important public safety matter numerically. The nature of the contribution of nuclear power plant radiation to cancer-related health effects should be clearly set forth in the safety goal policy statement, since it has such an important bearing on regulatory functions.

In the discussion of the plant performance guidelines the policy statement acknowledges the desirability of consequence mitigation for more serious core-melt accidents but fails to indicate the type of mitigation intended by the policy. There is a need to emphasize siting, emergency planning, and containment as major performance considerations. However, the intent of the policy concerning mitigation against the consequences of containment melt through, containment overpressure, and hydrogen combustion effects is not addressed. The policy should allow for consideration of these matters as requirements evolve. Probabilistic circumstances will have an important bearing on the value of mitigative features and a method of weighing costs and benefits probabilistically is needed.

The policy statement acknowledges the importance of sabotage, seismic events, and engineering and operating mistakes as matters needing consideration in the policy. These matters have to be controlled by regulatory functions implemented in a manner consistent with policy objectives. The policy statement should acknowledge the need for regulatory treatment of these matters if the goals are to be attained, but it should avoid claims that may not be demonstrable or even attainable. The policy should state clearly that licensed installations must have reserve capability that makes them tolerant of unintended mishaps arising from these uncertainties. It should also include a statement concerning surveillance, testing, review, and correction (including backfits) covered by regulatory functions intended to monitor problem areas related to these and other issues. The obligations of the regulated industry, the NRC, and other governmental agencies should all be identified. The policy could, however, point out that these risk control considerations also apply to other public structures, processes, and facilities and that nuclear plants are already required to be more vigilant concerning such matters than are other vulnerable facilities such as water supplies, public buildings, arsenals, and public gathering places of equivalent or greater public safety importance.

Additional Comments by ACRS Member M. Bender (Cont'd)

The policy statement should acknowledge the importance of public water, food supplies, public housing, and capital facilities that provide jobs as factors deserving consideration in the goals policy statement covering siting. As a practical matter, however, new considerations can be addressed only for plants not yet considered for licensing.

There would be great value in a safety goals policy that is easily understood by the public and that can be related to regulatory activities. Showing that the goals lead to improvements in failure trends, operator competence and plant reliability would have meaning and accomplishments would be measurable. The roles of the NRC, the regulated industry and other involved governmental and private agencies should be clearly stated. To be useful and usable, the policy should focus on these matters.

Additional Comments by ACRS Member H. W. Lewis

To begin with, I have to record that I do not endorse a number of points in this letter. However, I would like to join in commending the Commission for having taken a step in the direction of defining its position on "how safe is safe enough." In the end, the public, the Congress, and the Staff will benefit thereby. However, there is so much potential for misuse and misunderstanding in the goals as stated in NUREG-0880 that it is absolutely essential that a measured and thoughtful implementation plan be part of the package before useful comments can be made on the goals themselves. One must know not only how they are intended to be used, but how they can be misused. It would be an unfortunate and serious error to promulgate goals before these matters have been well scrubbed. I personally believe that PRA should not be part of a licensing package -- that is, that licensing should remain deterministic -- but that the regulations should be based upon careful threat analysis, increasingly using PRA. PRA is not yet mature enough to be debated in licensing hearings or in the courts. Increased internal use, however, will be beneficial to both reactor safety and to PRA.

The proposed goals illustrate (by failure) how difficult it is to decide "how safe is safe enough." An ideal society presumably makes social choices by comparing overall social cost (including risk) against overall social benefit (in this case, clean, cheap electricity). However, the NRC is neither a society nor ideal, and it is charged with assessing none of the benefits of nuclear power, and only one of the costs -- risk. Thus we find the proposed goals stated in terms of other accident threats (half of these

Additional Comments by ACRS Member H. W. Lewis (Cont'd)

are automobile accidents -- what has that to do with nuclear power? -- why not use measles?) or the risks associated with other means of making electricity (ignoring other elements of comparison -- must nuclear electricity be as safe as electricity made from a hypothetical fuel that is safe but produces epidemics of hives?). I believe that a safety goal should be an arbitrary expression of the NRC's assessment of a reasonable risk (overall risk) contribution to the societal cost of nuclear power. The body politic can then, taking into account the other factors, decide whether it is too high or too low. It would be used and developed internally to meet the overall objective -- the comparison with irrelevant things like automobile accidents and cigarette-induced cancer would not be part of the package, though they could, of course, be used for public education. Internally, there would, of course, have to be a great deal of PRA on elements of the plant, for the purpose of risk allocation through the regulatory process. At that level, flexible subgoals might be appropriate.

Given that general comment, there are just a few specifics about the proposal, that must be mentioned if the argument for delay does not prevail. These are by no means inclusive, but are again meant to demonstrate that there is still plenty of work to be done, and more can be added to the list.

First, the "most exposed individual". As I read the criterion, there is a positive incentive, in the event of an accident, to spread the radioactive effluent over as many people as possible, if by so doing one can reduce the maximum exposure to any single individual. I'm not convinced it is wise to provide that incentive, and believe that it is a serious mistake to open the door to risk manipulation that is inherent in this concept. I would prefer to aim at the greatest social good, and to make special provisions for the most exposed few.

Next, "risk aversion". It is true that large accidents attract public and press attention, but I see in that little reason for the NRC to be concerned with anything but overall risk to the public, unless it is demonstrated that large accidents do disproportionate damage. For example, if a meteorite were heading toward a town in which it would kill 100 people, should we break it up into fragments that kill 1000 people, but scattered across the country? I think not, but that is what "risk aversion" is all about. It is possible to think of arguments for the concentration of risk.

Finally, ALARA, which is the only place in which dollars appear. I find it odd that the same value is placed on a man-rem (in current dollars) as was the case when 10 CFR 50, Appendix I was issued some years ago. Has life cheapened at just the inflation rate, or has our increased understanding of

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the effects of low levels of radiation made it more benign at just the inflation rate? Or is it possible that it always was an arbitrary number? If so, why continue? Regardless of the number (which could, of course, be rationally derived, and have a cost-of-living adjustment), I believe the concept of ALARA is flawed, and that it is neither necessary nor appropriate if a well-considered overall quantitative safety goal is in place, and if a combination of PRA and expertise is in place to support that goal.

All of the above may be c[illegible]ed as a go-slowly-and-carefully recommendation, which includes proceeding apace with the internal use of PRA.