ACRS-2621
PDR 5/12/89

CERTIFIED

CERTIFIED COPY
DATE ISSUED:2/13/89

SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE
ON MECHANICAL COMPONENTS
JANUARY 27,1989
BETHESDA, MARYLAND

The ACRS Subcommittee on Mechanical Components met on January 27, 1989 in Room 422 at Bethesda, Maryland to review the NRC Staff's proposed resolution to Generic Issues 70, "Power Operated Relief Valve & Block Valve Reliability," and 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors."

Notice of the meeting was published in the Federal Register on January 17, 1989, (Attachment A). The schedule of items covered in the meeting is in Attachment B. A list of handouts kept with the office copy of the minutes is included in Attachment C. There were no written or oral statements received or presented from members of the public at the meeting. E. G. Igne was Cognizant ACRS Staff Member for the meeting.

Principal Attendees

ACRS

C. Michelson, Chairman
C. Wylie, Member
J. Carroll, Member
C. Siess, Member

NRC

K. Kniel
R. Baer
R. Kirkwood
E. Throm
F. Cherny
G. Mazetis

RS01
1/1

Others

W. McCaughey, BG&E
B. Gore, PNL
G. Murphy, ORNL

DESIGNATED ORIGINAL

Certified By EMB

CERTIFIED

Highlights

F. Cherny, RES, discussed the proposed resolution to Generic Issue 70, "PORV and Block Valve Reliability." F. Cherny briefly highlighted the history, described air actuated valves, and electrical actuated valves and discussed where/how PORVs and block valves are installed in PRWs in domestic nuclear power plants. He stated that these valves are mainly used in PWRs as pressurizer safety and relief valves.

Historically PORVs used during pre-1979 licensed plants were not safety grade. That is, valve operators and their electrical control systems were normally designed to non-safety related standards. However, the pressure-retaining elements of PORVs are within the reactor coolant pressure boundary (RCPB) and were constructed to the same codes and standards as those required for similar safety-related RCPB components. PORVs originally provided for operational flexibility. The role of PORVs has changed over a period of time and that they are now relied upon by many Westinghouse, B&W and CE plants to perform one or more of the following safety-related functions:

- Mitigating a design basis steam generator tube rupture accident.

- Provide low-temperature overpressure protection of the reactor vessel during start-up and cooldown.

- Venting of the reactor coolant system.

They also provide safety-related functions for events beyond the design basis, such as feed and bleed and ATWS mitigation.

Operating experience of PORVs and block valves is documented in an ORNL report NUREG/CR - 4692. The report reviewed events from 1971 to mid-1986. About 198 PORVs (mechanical, control or design) were found degraded or failed. 23% of the PORV mechanical events and 67% of the PORV control events were failures.

A PRA risk reduction study was done by BNL and reported in NUREG/CR - 4999. Results indicate that core melt frequencies attributable to PORV or block valve failures were found to be relatively insignificant and to represent only a very small fraction of total core melt frequency attributable to internal plant events. The NRC staff believes that the BNL results underestimates safety benefit that would be achieved by improving PORV and block valve reliability. However, BNL fault trees are dominated by operator error considerations, and it does not appear that results would have changed a great deal even if higher PORV failure rates were used. Consideration of feed and bleed (not within the scope of the BNL study, but evaluated as part of USI A-45) indicates a much greater safety importance of PORVs and block valves.

F. Cherny in the resolution of GI-70 recommends the following changes:

- For future PWR plants and plants currently under construction;

when PORVs and block valves are used for safety functions, these components should be classified as safety-related and a minimum of two PORVs and block valves be installed.

o For generating plants, a number of improvements (short of upgrading to fully safety-grade hardware) can increase reliability of PORVs and block valves and provide assurance that they will function as required. These improvements are as follows:

- Include PORVs and block valves within the scope of Appendix B.

- For most operating plants, modify the limiting conditions of operation of PORVs and block valves in the Technical Specifications for Modes 1, 2 and 3. The intent is to ensure plants that run with the block valves closed maintain electrical power to the block valves so they can be opened upon demand and not permit unlimited plant operation with PORVs and block valves inoperable for other reasons other than seat leakage.

- Use to the extent possible, more reliable PORV block valve designs that are resistant to failure.

In the supporting regulatory analysis, the NRC staff estimates that outage avoidance costs based on industry data reported by EPRI, would far exceed the cost of implementing the staff's recommendations (except for the last item above).

2. E. Throm, RES, discussed the proposed generic resolution GI-94, "Additional Low-Temperature Overpressure Protection (LTOP) for LWRs." This study was performed by Battelle Pacific Northwest Laboratory (PNL) under B. Gore. E. Throm stated that in 1979, multi-plant Action B-04 resulted in the imposition of new requirements for procedures and equipment to reduce the potential for cold overpressure events based on recommendations from USIA-26. Current staff requirements are in SRP 5.2.2 and Tech. Spec. for Overpressure protection. Basis for LTOP are found in GDC 15 and 31 of Appendix A and Appendix G to 10 CFR Part 50.

   Operating history indicates that only three (3) overpressure mitigating system failures occurred, and that the frequency of LTOP events is about the same as previously observed before 1980 or about 0.1/reactor-years. Also stated was that the likelihood of exceeding Appendix G pressure/temperature limit is about 1 in 10 per LTOP event.

A vessel through-wall crack probability analysis was performed. Results indicate that a core damage frequency (CDF) for Westinghouse and Combustion Engineering Nuclear Steam Supply System is about 3.24 x 10^{-6}/reactor-year. For a B&W plant, the mean CDF is negligible and no LTOP transient was observed to challenge the overpressure mitigating system.

Based on consequences analyses performed , the NRC staff stated that a Technical Specification only change is recommended to improve the overpressure mitigation system, and that no hardware changes are necessary.

Subcommittee Action

The subcommittee in its deliberation will recommend to the ACRS at the February meeting that the proposed Generic Issues 70 and 94 be accepted with clarifying comments for GI 70.

****** ****** ******

NOTE: A transcript of the meeting is available at the NRC Public Document, Gelman Bldg., 2120 "L" St. NW., Washington, D.C., Telephone (202) 634-3383 or can be purchased from Heritage Reporting Corporation, 1220 L Street, NW., Washington D.C. 20005, Telephone (202) 628-4888.

The proposed action is in accordance with the licensee's application dated August 30, 1988.

The Need for the Proposed Action

The proposed changes are needed so that the licensee can use higher enrichment fuel, and provide the flexibility of extending the fuel irradiation and permitting operation of longer fuel cycles.

Environmental Impacts of the Proposed Action

The staff has completed its evaluation of the proposed revisions to the Technical Specifications. The proposed revision would permit use of fuel enriched with Uranium 235 up to 4.85 weight percent. The safety considerations associated with reactor operation with higher enrichment and extended irradiation have been evaluated by the NRC staff. The staff has concluded that such changes would not adversely affect plant safety. The proposed changes have no adverse effect on the probability of any accident. The increased burnup may slightly change the mix of fission products that might be released in the event of a serious accident but such small changes would not significantly affect the consequences of serious accidents. No changes are being made in the types or amounts of any radiological effluents that may be released offsite during normal operation. There is also no significant increase in the allowable individual or cumulative occupational radiation exposure.

With regard to potential nonradiological impacts of reactor operation with higher enrichment and extended irradiation, the proposed changes to the TS involve systems located within the restricted area as defined in 10 CFR Part 20. They do not affect nonradiological plant effluents and have no other environmental impact.

The environmental impacts of transportation resulting from the use of higher enrichment fuel and extended irradiation have been discussed in the staff assessment entitled, "NRC Assessment of the Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation," dated July 7, 1988 (53 FR 30355). As indicated therein, the environmental cost contribution of the proposed increase in the fuel enrichment and irradiation limits are either unchanged or may in fact be reduced from those summarized in Table S-4 as set forth in 10 CFR 51.42(c).

Therefore, the staff concludes that there are no significant radiological or nonradiological environmental impacts associated with the proposed amendment.

Alternative to the Proposed Action

Since the staff concluded that there are no significant environmental effects that would result from the proposed action, any alternatives with equal or greater environmental impacts need not be evaluated.

The principal alternative would be to deny the requested amendment. This would not reduce environmental impacts of plant operation and would result in reduced operational flexibility.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the "Final Environmental Statement related to the operation of the Beaver Valley Power Station, Unit 2," dated September 1985.

Agencies and Persons Consulted

The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

Finding of No Significant Impact

The staff has determined not to prepare an environmental impact statement for the proposed license amendment.

Based upon the foregoing environmental assessment, the staff concluded that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the application for amendment dated August 30, 1988, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

Dated at Rockville, Maryland, this 9th day of January, 1989.

For the Nuclear Regulatory Commission.

Peter S. Tam,

Acting Director, Project Directorate I-4, Division of Reactor Projects I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 89-969 Filed 1-13-89; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Meeting of the Subcommittee on Human Factors; Meeting

The ACRS Subcommittee on Human Factors will a meeting on January 26, 1989, Room P-422, 7920 Norfolk Avenue, Bethesda, MD.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, January 26, 1989—8:30 a.m. until the conclusion of business

The Subcommittee will review the Human Factors Research Program Plan.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS staff member named below as far in advance as is practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant ACRS staff member, Mr. Herman Alderman (telephone 301/492-7750) between 7:30 a.m. and 4:30 p.m. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Date: January 9, 1989.

Morton W. Libarkin,

Assistant Executive Director for Project Review.

[FR Doc. 89-966 Filed 1-13-89; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Subcommittee on Mechanical Components; Meeting

The ACRS Subcommittee on Mechanical Components will hold a meeting on January 27, 1989, Room P-422, 7920 Norfolk Avenue, Bethesda, MD.

The entire meeting will be open to public attendance.

The agenda for subject meeting shall be as follows:

Friday, January 27, 1989—2:00 p.m. until the conclusion of business

The Subcommittee will review the proposed resolution of Generic Issues 70, "PORV Reliability," and 94, "Low Temperature Over-Pressure Protection," and other related matters.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the ACRS staff member identified below as far in advance as practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant ACRS staff member, Mr. Elpidio Igne (telephone 301/492-8192) between 7:30 a.m. and 4:15 p.m. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Date: January 9, 1989.

Morton W. Libarkin,

Assistant Executive Director for Project Review.

[FR Doc. 89-987 Filed 1-13-89; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Subcommittee on Auxiliary and Secondary Systems; Meeting

The ACRS Subcommittee on Auxiliary and Secondary Systems will hold a meeting on January 27, 1989, Room P-422, 7920 Norfolk Avenue, Bethesda, MD.

The entire meeting will be open to public attendance.

The age[illegible]a for the subject meeting will be as follows:

Friday, January 27, 1989—8:30 a.m. until 1:00 p.m.

The Subcommittee will review the adequacy of the proposed Staff's plans to implement the recommendations resulting from the Fire Risk Scoping Study.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS Staff member named below as far in advance as is practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant ACRS Staff member, Mr. Sam Duraiswamy (telephone 301/492-9522) between 7:30 a.m. and 4:15 p.m. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Date: January 9, 1989.

Morton W. Libarkin,

Executive Director for Project Review.

[FR Doc. 89-988 Filed 1-13-89; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-315]

Indiana Michigan Power Co.; Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing

The United States Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-58 issued to the Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook Nuclear Plant, Unit No. 1 (the facility), located in Berrien County, Michigan.

In accordance with the licensee's application for amendment dated August 9, 1988, the amendment would allow a one-time extension of the surveillance intervals for certain surveillances normally performed with the unit shutdown. The extensions involve:

1. Ice basket weighing;
2. Ice condenser flow passage inspections;
3. Ice condenser inlet door testing; and
4. Resistance temperature detector calibrations.

Prior to issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commision has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission's evaluation of the proposed amendment with respect to the above criteria is as follows:

Criterion 1

The period of time the surveillance intervals will be exended is very brief. The surveillances for which extensions are requested fall due from February 26, 1989, through March 26, 1989. Extensions are necessary only until the Cycle 10-11 refueling outage, which is scheduled to begin in March 1989. The licensee has reviewed previous surveillance test results for the affected equipment and has found nothing that would indicate

TENTATIVE SCHEDULE

ACRS SUBCOMMITTEE ON MECHANICAL COMPONENTS MEETING
JANUARY 27, 1989
7920 NORFOLK AVENUE, BETHESDA, MARYLAND, ROOM P-422

8:30 - 8:40 a.m.	Opening Remarks C. Michelson, Chairman
8:40 - 10:26 ~~10:00~~ a.m.	Generic Issue 70; "Power Operated Relief Valve & Block Valve Reliability" R. Kirkwood, RES
10:26 - 10:40 ~~10:00 - 10~~:15 a.m.	--------- BREAK ---------
10:40 - 12:00 ~~10:15 - 11:30~~ a.m.	General Issue 94; "Additional Low Temperature Overpressure Protection for Light Water Reactors" E. Throm, RES
12:00 - 12:05 ~~11:30 - 12:00~~ NOON	Subcommittee Discussion and Adjournment

ATTACHMENT C

LIST OF HANDOUTS

1. Presentation to ACRS Subcommittee, Mechanical Components Generic Issue 70 PORV and Block Valve Reliability by Senior Task Mgr. R. Kirkwood, EIB and Presentation by F. Cherny, Section Leader EIB

2. Generic Issue 94 "Additional Low-Temperature Overpressure Protection (LTOP) for LWRs" by Edward D. Throm, Senior Task Manager