REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:7903270461 DOC.DATE: 79/03/19 NOTARIZED: NO DOCKET #
FACIL:50-315 DONALD C. COOK NUCLEAR POWER PLANT, UNIT 1, INDIANA & 05000315
50-316 DONALD C. COOK NUCLEAR POWER PLANT, UNIT 2, INDIANA & 05000316
AUTH.NAME AUTHOR AFFILIATION
DISBROW,R.E. INDIANA & MICHIGAN POWER CO.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. OFFICE OF NUCLEAR REACTOR REGULATION

SUBJECT: REQUESTS REVISIONS TO ROD POSITION INDICATOR TECH SPECS TO REMOVE REFERENCES TO PART LENGTH CONTROL RODS & TO CHANGE REPORTING REQUIREMENTS.W/ENCL ADDL DETAILS & REVISED TECH SPEC PAGES.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16
TITLE: GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LIC

NOTES: w/check $1,600.00

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC ORB #1	7	7			
INTERNAL:	01 REG FILE	1	1	02 NRC PDR	1	1
	12 I&E	2	2	14 TA/EDO	1	1
	15 CORE PERF BR	1	1	16 AD SYS/PROJ	1	1
	17 ENGR BR	1	1	18 REAC SFTY BR	1	1
	19 PLANT SYS BR	1	1	20 EEB	1	1
	21 EFLT TRT SYS	1	1	22 BRINKMAN	1	1
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	23 ACRS	16	16			

MAR 28 1979

MAR 28 1979

MA 4
BD

TOTAL NUMBER OF COPIES REQUIRED: LTTR 38 ENCL 38

INDIANA & MICHIGAN POWER COMPANY
P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

March 19, 1979
AEP:NRC:00145

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

This letter requests revisions to the Rod Position Indicator (RPI) System Technical Specifications for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. Appendix A Technical Specifications 3/4.1.3.1 and 3/4.1.3.2 have been revised to indicate the following:

(1) References to part-length control rods have been removed.

(2) The Technical Specification Reporting Requirements for the RPI System have been changed to allow for the submittal of a Licensee Event Report (LER) only when the actual, as opposed to the indicated, rod misalignment is greater than ± 12 steps from the bank demand position. Please note, though, that the surveillance requirements for the RPI System have not been lessened. Only the resulting reporting requirements are being changed.

More detailed discussions of the above Technical Specification change request and a copy of the corresponding revised pages are contained in the Attachments to this letter. The revisions to the RPI Reporting requirements contained therein have been discussed with members of your staff and found acceptable. AEPSC respectfully requests that this Technical Specification change request package be processed in an expeditious manner to preclude the future submittal of LERs due to false indication of the RPI system.

A001
S
1/1
w/check
$ 11,600.00

[illegible]

This Technical Specification change request has been reviewed by the PNSRC and the AEPSC NSDRC, in accordance with the appropriate provisions of our Technical Specifications. The result of these reviews indicates that in no instance will the subject Technical Specification change adversely affect the health and safety of the public.

This Technical Specification change request is considered to be a Class II License Amendment as per the provisions of 10 CFR 170.22. As required by Part 170 Subsection 22 a check for $1,600.00 accompanies this submittal.

Very truly yours,

R. E. Disbrow
Vice President

RED:em

Sworn and subscribed to before me this 19th day of March, 1979 in New York County, New York

Gregory M. Gurican
Notary Public

GREGORY M. GURICAN
Notary Public, State of New York
No. 31-4643431
Qualified in New York County
Commission Expires March 30, 1981

cc: R. C. Callen
G. Charnoff
R. Walsh
P. W. Steketee
R. J. Vollen
D. V. Shaller-Bridgman
R. W. Jurgensen

ATTACHMENT "A" TO AEP:NRC:00145

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

Technical Specification Paragraph 3.1.3.1

Basis for Technical Specification Change:

This proposed revision deletes all reference to "(indicated position)" which previously appeared in paragraph 3.1.3.1 and Action Statements (b) and (c).

The purpose for the revision is to allow alternate methods for physical rod position verification without altering the intent of the Specification.

As the intent of the Specification is primarily concerned with actual rather then indicated rod position, two alternate methods for position verification are available for use in addition to the position indication system. The two alternate methods are (a) rod detector (L.V.D.T.) secondary coil voltage measurements and (b) the movable incore detectors.

The primary source of rod position determination will continue to be the rod position indication system although with the alternate method available in the event of an indicator in excess of the ± 12 step requirement, the surveillance requirements and specification intent can still be satisfied.

It should also be noted that Specification 3.1.3.2 imposes sufficient requirements to insure both the availability and accuracy of the position indicators.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, which are inserted in the core, shall be OPERABLE and positioned within ± 12 steps of their bank demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length Rod(s) inoperable or misaligned from the bank demand position by more than ± 12 steps, be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable or misaligned from its group step counter demand height by more than ± 12 steps, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied.
POWER OPERATION may then continue provided that:

a) An analysis of the potential ejected rod worth is performed within 3 days and the rod worth is determined to be ≤ 0.75% Δk at zero power and ≤ 0.38% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least one per 12 hours, and

* See Special Test Exceptions 3.10.2 and 3.10.4

LIMITING CONDITION FOR OPERATION (Continued)

c) The THERMAL POWER level is reduced to ≤ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER, or

d) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

Technical Specification Paragraph 3.1.3.2

Basis for Technical Specification Change:

This proposed revision adds Action Statement (a1) whose purpose is to:

1) Allow the use of rod detector (L.V. D. T.) secondary coil voltage measurements to verify proper rod alignment with bank demand in the event of indicator inoperability;

2) Allow eight hours for the adjustment, repair or replacement of the inoperable indicator and/or associated circuitry;

3) Exempt occurrences of this type from the 30 day NRC reporting requirements stipulated in Technical Specification 6.9.1.9 provided that all actions stated are performed within the specified time limit and the affected rod is properly positioned with 12 ± steps of the bank demand position.

<u>Basis</u>

Both the reporting exemption and the eight hour allowable repair time limit are based on the knowledge that the actual position of the non-indicating rod has been verified to be positioned within ± 12 steps from bank demand and that the indicator itself has been declared inoperable.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

1. Determine the actual position of the rod(s) not indicating to be within ± 12 steps of bank demand position by Rod Detector (L.V.D.T.) Secondary Coil Voltage Measurements and return the inoperable indicating channel(s) to OPERABLE status within 8 hours (the provisions of Specification 6.9.1.9 are not applicable), or,

2. Determine the position of the rod(s) not indicating to be within ± 12 steps of bank demand position indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

3. Reduce THERMAL POWER TO <50% OF RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to ≤ 50% RATED THERMAL POWER within 8 hours.

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

ATTACHMENT "B" TO AEP:NRC:00145

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

Technical Specification Paragraph 3.1.3.1

Basis for Technical Specification Change:

This proposed revision deletes all reference to "(indicated position)" which previously appeared in paragraph 3.1.3.1 and Action Statements (b) and (c).

The purpose for the revision is to allow alternate methods for physical rod position verification without altering the intent of the Specification.

As the intent of the Specification is primarily concerned with actual rather then indicated rod position, two alternate methods for position verification are available for use in addition to the position indication system. The two alternate methods are (a) rod detector (L.V.D.T.) secondary coil voltage measurements and (b) the movable incore detectors.

The primary source of rod position determination will continue to be the rod position indication system although with the alternate method available in the event of an indicator in excess of the ± 12 step requirement, the surveillance requirements and specification intent can still be satisfied.

It should also be noted that Specification 3.1.3.2 imposes sufficient requirements to insure both the availability and accuracy of the position indicators.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core, shall be OPERABLE and positioned within ± 12 steps of their group step counter damand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps, be in HOT STANDBY within 6 hours.

c. With one full rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps, POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

*See Special Test Exceptions 3.10.2 and 3.10.3

LIMITING CONDITION FOR OPERATION (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) Either the THERMAL POWER level is reduced to $\leq$ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verifying the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

Technical Specification Paragraph 3.1.3.2

Basis for Technical Specification Change:

This proposed revision adds Action Statement (a1) whose purpose is to:

1) Allow the use of rod detector (L.V. D. T.) secondary coil voltage measurements to verify proper rod alignment with bank demand in the event of indicator inoperability;

2) Allow eight hours for the adjustment, repair or replacement of the inoperable indicator and/or associated circuitry;

3) Exempt occurrences of this type from the 30 day NRC reporting requirements stipulated in Technical Specification 6.9.1.9 provided that all actions stated are performed within the specified time limit and the affected rod is properly positioned with 12 $\pm$ steps of the bank demand position.

Basis

Both the reporting exemption and the eight hour allowable repair time limit are based on the knowledge that the actual position of the non-indicating rod has been verified to be positioned within $\pm$ 12 steps from bank demand and that the indicator itself has been declared inoperable.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

1. Determine the actual position of the rod(s) not indicating to be within ± 12 steps of bank demand position by Rod Detector (L.V.D.T.) Secondary Coil Voltage Measurements and return the inoperable indicating channel(s) to OPERABLE status within 8 hours (the provisions of Specification 4.1.1.1 are not applicable), or,

2. Determine the position of the rod(s) not indicating to be within ± 12 steps of bank demand position indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

3. Reduce THERMAL POWER TO <50% OF RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to ≤ 50% RATED THERMAL POWER within 8 hours.

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

50-316

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 15, 1979

3/15/79

ALL POWER REACTOR LICENSEES

Gentlemen:

On September 14, 1978, the Nuclear Regulatory Commission established a new Pipe Crack Study Group which was to evaluate recent pipe and safe end cracking experience relative to previous staff conclusions and recommendations. The bases for establishing the new Study Group were (1) the discovery of cracks in the inner surface of large-diameter austenitic stainless steel piping (recirculation lines) in a BWR and (2) questions concerning the capability of ultrasonic detection methods to detect small cracks.

The new PCSG reviewed existing information that either was contained in written records or had been collected through meetings in this country and in foreign countries. The review was in the context of changes occurring since the preparation by the original Pipe Cracking Study Group of NUREG-75/067 "Technical Report--Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants". The conclusions and recommendations of the new Pipe Crack Study Group are presented in the enclosed "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants", NUREG-0531. This report is for your information and comment. Also enclosed is a copy of a related Federal Register Notice.

The NRC staff will review the Study Group report and its conclusions/recommendations and any comments received about the report. Following this review, the staff will decide what further actions, if any, are required for the licensing and operation of reactors.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NUREG-0531
2. Notice

MA 4

GD

[7590-01-M]

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Request for public comment on NUREG-0531 "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants" February 1979.

SUMMARY: On September 14, 1978, the Nuclear Regulatory Commission established a new Pipe Crack Study Group. The Group was to evaluate recent pipe and safe end cracking experience relative to previous staff conclusions and recommendations. The NRC seeks public comment on the report which summarizes the Group's review and conclusions.

DATES: The public comment period expires May 15, 1979.

FOR FURTHER INFORMATION CONTACT:

Darrell G. Eisenhut, Deputy Director for Operating Reactors, Division of Operating Reactors, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. (Phone: 301-492-7221)

SUPPLEMENTARY INFORMATION: In 1975, a Pipe Cracking Study Group was established by the United States Nuclear Regulatory Commission (USNRC) to review intergranular stress-corrosion cracking (IGSCC) in Boiling Water Reactors (BWRs). The Group reported its findings concerning stress-corrosion cracking in by-pass lines and core spray piping of austenitic stainless steel in a report, *Technical Report—Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants* (NUREG-75/067).

During 1978, IGSCC was reported for the first time in large-diameter piping in a BWR. This discovery, together with questions concerning the capability of ultrasonic detection methods to detect small cracks, led to the formation of a new Pipe Crack Study Group (PCSG) by USNRC on September 14, 1978.

The charter of the new PCSG was to specifically address the five following questions:

"1. The significance of the cracks discovered in large-diameter pipes relative to the conclusions and recommendations set forth in the referenced report (NUREG-75/067) and its implementation document, NUREG-0313;

2. Resolution of the concerns raised over the ability to use ultrasonic techniques to detect cracks in austenitic stainless steel;

3. The significance of cracks found in large-diameter sensitized safe ends and any recommendations regarding the current NRC program for dealing with this matter;

4. The potential for stress corrosion cracking in PWRs;

5. Examine the significance of cracking in the Inconel safe ends that has been experienced at the Duane Arnold Operating Facility, and develop any recommendations regarding NRC actions taken or to be taken."

The PCSG limited the scope of the study to BWR and PWR piping and safe ends attached to the reactor pressure vessel. The PCSG reviewed existing information—either that contained in written records or that collected through meetings in this country and in foreign countries. The specific areas considered are presented in the chapters of this report:

- BWR Cracking Experience and Corrective Actions
- PWR Cracking Experience and Corrective Actions
- Metallurgy Associated with Pipe Cracking
- Reactor Coolant Chemistry
- Pipe Configuration and Stress Levels
- Duane Arnold Safe-End Cracking
- Methods of Detecting Cracks
- Significance of Cracks
- Recent Development Relevant to Control and Detection of IGSCC

The review of these topics in the context of changes occurring since the preparation of NUREG-75/067 led to the preparation of specific conclusions and recommendations relevant to the current status of IGSCC, the significance of the problem, and the reliability of detection and measures available to correct or minimize IGSCC in existing and future plants. These conclusions and recommendations are presented in the newly issued PCSG report.

The NRC staff will review the Study Group report and its conclusions/recommendations and the public comments received during this comment period. Following this review, the staff will decide what further actions, if any, are required for the licensing and operation of reactors.

Requests for a single copy of the report should be made in writing to U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

Comments on this report should be sent to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Duputy Director, Division of Operating Reactors. The comment period expires May 15, 1979. Copies of all comments received will be available for examination in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.

Dated at Bethesda, Md., this 6th day of March, 1979.

For the Nuclear Regulatory Commission.

VICTOR STELLO, Jr.,
Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation.

[FR Doc. 79-7705 Filed 3-12-79; 8:45 am]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

U.S. MAIL

Mr. John Tillinghast
Indiana and Michigan Electric Company
Indiana and Michigan Power Company

cc: Mr. Robert W. Jurgensen
Chief Nuclear Engineer
American Electric Power
Service Corporation
2 Broadway
New York, New York 10004

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

David Dinsmore Comey
Executive Director
Citizens for a Better Environment
59 East Van Buren Street
Chicago, Illinois 60605

Maude Reston Palenske Memorial
Library
500 Market Street
St. Joseph, Michigan 49085

Mr. D. Shaller, Plant Manager
Donald C. Cook Nuclear Plant
P. O. Box 458
Bridgman, Michigan 49106

Kenneth R. Baker
2874 Robin Hood Drive
Stevensville, Michigan 49127