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Docket No.: 50-271

MAR. 11 1976

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 20 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications and is based on our letters to you dated September 19, 1975 and December 19, 1975.

This amendment revises the Technical Specifications to (1) add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod drive mechanism collet housing failures and (2) require increased control rod surveillance when the possibility of a control rod drive mechanism collet housing failure exists.

We have evaluated the potential for environmental impact of plant operation in accordance with the enclosed amendment and have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level, and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR § 51.5(d)(4) that an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by this action.

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

A copy of the related Federal Register Notice is also enclosed. Our Safety Evaluation relating to this action was forwarded to you with our letter dated September 19, 1975.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 20 to
   License No. DPR-28
2. Federal Register Notice

cc w/enclosures: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR PRB	OELD DL	C-ORB#4:DOR	AD-EOR:DOR KRG	
SURNAME	RIngram:rm	PDiBenedetto:rm	D SWANSON	RReid	KRGoller	
DATE	3/1/76	3/2/76	3/9/76	3/2/76	3/10/76	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   B. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   C. The facility will operate in conformity with the provisions of the Atomic Energy Act of 1954, as amended, and the rules and regulations of the Commission; and

   D. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR. 11 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Delete existing pages 68, 69, and 75 of the Technical Specifications and insert the attached revised pages. The changed areas on the revised pages are shown by marginal lines.

3.3 CONTROL ROD SYSTEM

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operation cycle with the highest worth, operable control rod in its fully withdrawn position and all other operable rods inserted.

2. Reactivity margin - inoperable control rods

Control rod driven which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. The control rod directional control valves for inoperable control rods shall be disarmed electrically except for control rods which are inoperable because of scram times

4.3 CONTROL ROD SYSTEM

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate a shutdown margin of 0.25 per cent Δk at any time in the subsequent fuel cycle with the highest worth operable control rod fully withdrawn and all other operable rods inserted.

2. Reactivity margin - inoperable control rods

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be greater than those specified in Specification 3.3.C. In no case shall the number of inoperable rods which are not fully inserted be greater than six during power operation.

B. Control Rods

1. Each control rod shall be either coupled to its drive or placed in the inserted position and its directional valves disarmed electrically. When removing up to one control rod drive per quadrant for inspection and the reactor is in the refueling mode, this requirement does not apply.

completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

B. Control Rods

1. The coupling integrity shall be verified:

    (a) When a rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernable response of the nuclear instrumentation; however, for initial rods when response is not discernable, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response; and

    (b) When a rod is fully withdrawn, observe that the rod does not go to the over-travel position. Prior to startup following a refueling outage, each rod shall be fully withdrawn continuously to observe that the rate of withdrawal is proper and that the rod does not go to the over-travel position. Following uncoupling, each control rod drive and blade shall be tested to verify positive coupling and the results of each test shall be recorded. This test shall consist of checking the operability of the over-travel indicator circuit prior to coupling by withdrawing the drive and observing the over-travel light. The drive and blade shall then be immediately coupled and fully withdrawn. The position and over-travel lights shall be observed.

A. Reactivity Limitations

1. Reactivity margin - core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. At each refueling the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk with the highest worth control rod fully withdrawn and all others inserted. The value of R in %Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check. R must be a positive quanity or zero.

The 0.25% Δk in the expression, R + 0.25% Δk, is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the highest worth rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% Δk in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the highest worth rod fully withdrawn but at least a R + 0.25% Δk margin. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core.

2. Reactivity margin - inoperable control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If a rod is disarmed electrically, its position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods, assuming the highest worth, operable control rod does rod insert. An allowable pattern for control rods valved out of service will be available to the reactor operator. The number of rods permitted to be inoperable could be many more than the six allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than six could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housing, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-28 issued to the Vermont Yankee Nuclear Power Corporation (the licensee), which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility), located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to (1) add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod drive mechanism collet housing failures and (2) require increased control rod surveillance when the possibility of a control rod drive mechanism collet housing failure exists.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on January 7, 1976, (41 F.R. 1333). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the Commission's letters to Vermont Yankee Nuclear Power Corporation dated September 19, 1975, and December 19, 1975, (2) Amendment No. 20 to License No. DPR-28, and (3) the Commission's related Safety Evaluation issued on September 19, 1975. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Brooks Memorial Library, 244 Main Street, Brattleboro, Vermont.

A single copy of items (1) through (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 11th day of March 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

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

Treanor w/ letter
dtd. 9-19-75

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NO. DPR-28

AND

CHANGES TO THE TECHNICAL SPECIFICATIONS

INOPERABLE CONTROL ROD LIMITATIONS

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

On June 27, 1975, Commonwealth Edison Company (CE) informed NRC that cracks had been discovered on the outside surface of the collet housings of four control rod drives at Dresden Unit 3[1]. The cracks were discovered while performing maintenance of the control rod drives; the reactor was shutdown for refueling and maintenance. In a letter dated July 3, 1975, CE informed us that if the cracks propagated until the collet housing failed, the affected control rod could not be moved[2]. In a meeting with representatives of General Electric (GE) and CE we were advised that further inspections revealed cracks in 19 of the 52 Dresden 3 control rod drives inspected, in one spare Dresden 2 control rod drive, in one Vermont Yankee spare control rod drive and in two GE test drives[3]. In a report dated July 30, 1975, after additional rod drives were inspected, CE stated that cracks had been found in 24 of 65 drives inspected[4]. Recently, the Tennessee Valley Authority reported that cracks were found in the collet housing of

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(1) Telegram to J. Keppler, Region III of the NRC, June 27, 1975, Docket No. 50-249.

(2) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 3, 1975, Docket No. 50-249.

(3) Memo from L. N. Olshan, Division of Technical Review (DTR) to T. M. Novak, DTR, "Meeting on Cracks Found in Dresden 3 Control Rod Drive Collet Retainer Tubes," July 18, 1975.

(4) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 30, 1975, Docket No. 50-249.

seven of nineteen drives inspected at Browns Ferry 1 and Vermont Yankee found cracks in the collet housing of 4 of 10 control rod drives inspected. Because a number of control rod drives have been affected, because complete failure of the drive collet housing could prevent scram of the affected rod, and because we do not consider existing license requirements adequate in view of the collet housing cracks experienced, we have concluded that the Technical Specifications should be changed for those reactors with control rod drive designs susceptible to collet housing cracks. The change should assure that reactors which could be affected would not be operated for extended periods of time with a control rod which cannot be moved.

DESCRIPTION

The control rod drive is a hydraulically operated unit made up primarily of pistons, cylinders and a locking mechanism to hold the movable part of the drive at the desired position. The movable part of the drive includes an index tube with circumferential grooves located six inches apart. The collet assembly which serves as the index tube locking mechanism contains fingers which engage a groove in the index tube when the drive is locked in position. In addition to the collet, the collet assembly includes a return spring, a guide cap, a collet retainer tube (collet housing) and collet piston seals. The collet housing surrounds the collet and spring assembly. The collet housing is a cylinder with an upper section of wall thickness 0.1 inches and a lower section with a wall thickness of about 0.3 inches. The cracks occurred on the outer surface of the upper thin walled section near the change in wall thickness.

1. Consequences of Cracking

The lower edges of the grooves in the index tube are tapered, allowing index tube insertion without mechanically opening the collet fingers, as they can easily spring outward. If the collet housing were to fail completely at the reported crack location, the coil collet spring could force the upper part of the collet housing and spring retainer upward, to a location where the spring and spring retainer would be adjacent to the collet fingers. The clearance between the collet fingers and the spring when in this location will not permit the collet fingers to spring out of the index tube groove. This would lock the index tube in this position so that the control rod could not be inserted or withdrawn.

The failure of up to eight control rods to operate has previously been evaluated and the Technical Specifications presently allow up to eight rods to be inoperable. If more than eight rods are inoperable or if the scram reactivity rate is too small or if shutdown reactivity requirements are not met, the existing Technical Specifications require the reactor to be brought to a cold shutdown condition. Reactor power operation with up to eight rods inoperable would not involve a new hazards consideration nor would it endanger the health and safety of the public.

2. Probable Cause of Cracking

The cause of the cracking appears to be a combination of thermal cycling and intergranular stress corrosion cracking. The thermal cycling results from insertion and scram movements. During these movements hot reactor water is forced down along the outside of the collet housing, while cool water is flowing up the inside and out of flow holes in the housing. These thermal cycles are severe enough to yield the material, leaving a high residual tensile stress on the outer surface.

The collet housing material is type 304 austenitic stainless steel. The lower portion of the collet housing has a thicker wall and its inner surface is nitrided for wear resistance. In 1960-61, similar drives using high hardness 17-4 PH material for index tubes and other parts were found to have developed cracks. The problem caused GE to switch to nitrided stainless steel. The nitriding process involves a heat treatment in the 1050 F to 1100 F range, which sensitizes the entire collet housing, making it susceptible to oxygen stress corrosion cracking.

The cooling water used in the drives is aerated water. This water contains sufficient oxygen for stress corrosion to occur in the sensitized material if it is subjected to the proper combination of high stresses and elevated temperatures.

We believe that the cracking is caused by a combination of thermal fatigue and stress corrosion. GE has determined that both full stroke insertion and scram will cause high thermal stress. The cracks are completely intergranular and extensively branched, indicating that corrosion is a major factor. The type of thermal cycling, plus the buildup of corrosion products in the cracks between cycles probably results in a ratcheting action. This is also indicated by the "bulged" appearance of the cracks on the OD.

3. Probability of Early Failure

We believe that the cracking is progressive and is cycle dependent. Although the details of the cracking process are still not clear, we have not identified any mechanism that would cause rapid cracking with progression to complete circumferential failure.

The axial loads on the housings are very low at all times so that through wall cracks would have to progress at least 90% around the circumference before there would be concern about a circumferential failure. Although one housing at Dresden 3 had three cracks which nearly joined around the circumference, no cracks at Dresden 3 were through wall and none of the housings examined approached the degree of cracking necessary for failure. The collet housing has three flow holes in the thin section equally spaced around the circumference. The observed cracks have been confined primarily to the areas below and between the holes and near the area where the wall thickness of the collet housing changes. Since all the cracks except those located at the change in wall thickness are fairly shallow and since those at the change in wall thickness are largely confined to the circumferential area between holes, the net strength of the cracked housings is still far greater than necessary to perform their function.

A test drive at GE that had experienced over 4000 scram cycles had a more extensive developed crack pattern. Although the satisfactory experience with this cracked test housing is encouraging, its performance may not be correlated directly to that of drives in service, as this test drive was subjected to lower temperatures, and possibly less severe thermal cycles than could be encountered in actual service. The cracks were first noticed on the test drive after about 2000 cycles - many more cycles than the cracked housings at Dresden 3 had experienced.

The chance that a large number of collet housing would fail completely at about the same time is very remote. This is primarily true because the distributions of failures by cracking mechanisms such as stress corrosion and fatigue are not linear functions. That is, failure is a function of log time or log cycles. Distribution of failures of similar specimens generally follow a log normal pattern, with one to two orders of magnitude in time or cycles between failures of the first and failures of the last specimen. As no collet housing has yet failed, we are confident that there would be very few, if any, failures during the next time period corresponding to the total service life to date.

4. Changes to Technical Specifications

Existing limiting conditions of operation allow operation to continue with up to eight inoperable control rods. Existing surveillance requirements specify that daily surveillance of the condition of all fully or partially withdrawn rods would not have to begin until three rods are found inoperable. We do not consider that these existing limiting conditions of operation and surveillance requirements sufficiently limit the possibility of operating for an extended period of time with a number of rod drive mechanisms which cannot be moved. We have therefore concluded that the Technical Specifications should be changed as discussed below.

(a) One stuck control rod does not create a significant safety concern. However, if a rod cannot be moved and the cause of the failure cannot be determined, the rod could have a failed collet housing. A potentially failed collet housing would be indicative of a problem which could eventually affect the scram capability of more than one control rod. Since the cracks appear to be of a type which propagate slowly, it is highly unlikely that a second control rod would experience a failed collet housing within a short period of time after the first failure. Therefore, a period of time of 48 hours can be allowed to determine the cause of failure. This period is considered long enough to determine if the cause of failure is not in the drive mechanism, yet short enough to be reasonably assured that a second collet failure does not occur. Therefore Section 3.3.A.2 (Reactivity Margin - Inoperable Control Rods) should be expanded to require that if a control rod cannot be moved during normal operation, testing or scram, the reactor shall be shutdown within 48 hours if the reason that it cannot be moved cannot be shown to be due to causes other than a failed collet housing.

(b) If a control rod drive cannot be moved, the cause of the stuck rod might be a problem affecting other rods. To ensure prompt detection of any additional control rod drive failures which could prevent movement, Section 4.3.A.2 should be expanded to require surveillance every 24 hours of all partially and fully withdrawn rods if one rod drive is found to be stuck.

Until permanent corrective measures are taken to resolve the potential for stuck control rods due to failed collet housings, we believe that these additional specifications provide reasonable assurance that an unacceptable number of control rod collet housing will not fail during operation. Upon completion of the investigations being performed by GE, additional corrective actions may permit revision of these requirements.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

SEP 19 1975