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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

Introduction

By letters dated October 29, 1976, January 26, 1977, and February 23, 1977, Metropolitan Edison Company (the licensee) requested several amendments to Appendix A to Facility Operating License DPR-50 for Three Mile Island Nuclear Station Unit No. 1 (TMI-1). The change requested by the licensee's letter of October 29, 1976, would revise the TMI-1 Technical Specifications to permit irradiation of the TMI-1 reactor vessel material surveillance specimens in the Three Mile Island Nuclear Station Unit No. 2 (TMI-2) reactor vessel pursuant to the provisions of 10 CFR 50, Appendix H, Section II.C.4. By letter dated December 17, 1976, we requested additional information concerning this proposed amendment. This information was furnished by the licensee in letters dated December 29, 1976, and January 20, 1977.

The change requested by the licensee's letter of January 26, 1977, would revise the TMI-1 Technical Specifications to reflect plant operating limitations for the fuel loading to be used during Cycle 3. By letter dated March 23, 1977, we requested additional information concerning this proposed amendment. This information was furnished by the licensee's letter of March 31, 1977.

The change requested by the licensee's letter of February 23, 1977, would revise the TMI-1 Technical Specifications to update the reactor coolant system pressure limits during system heatup and cooldown. In support of this request the licensee also submitted Babcock and Wilcox Report "Analysis of Capsule TMI-1E from Metropolitan Edison Company Three Mile Island Nuclear Station Unit 1," BAW-1439 (January 1977).

Discussion and Evaluation

1. Reactor Vessel Material Surveillance Program

The original TMI-1 design included three reactor vessel surveillance specimen holder tubes (SSHTs) located near the reactor inside vessel wall. Each of these SSHTs housed two capsules containing reactor vessel surveillance specimens. During a refueling outage in the Spring of 1976 it was discovered that the SSHTs had suffered severe damage. To prevent further damage, all surveillance capsules and all parts of the SSHTs that had failed or were deemed likely to fail were removed from the vessel.

The NRC granted the licensee an exemption which permitted operation of TMI-1 in the following Cycle (Cycle 2) without the surveillance specimens present. This exemption was granted on the basis that the surveillance specimens had already received a neutron exposure in excess of that which the reactor vessel would receive by the end of Cycle 2. At the time the exemption was granted it was expected that SSHTs of improved design would be reinstalled in TMI-1.

Since the discovery of the damage to the SSHTs, Babcock & Wilcox Company (B&W), the reactor supplier, has undertaken the design, manufacture and testing of an improved SSHT. SSHTs of this improved design are presently installed in Davis-Besse Unit No. 1, Crystal River Unit No. 3 and Three Mile Island Unit No. 2 (TMI-2). All three of these plants have reactors supplied by B&W and all are in the process of beginning initial operation within the next few months. In addition, all of these reactors are of the same basic B&W 177 fuel assembly vessel design as TMI-1. The acceptability of the redesigned SSHTs has been demonstrated by a test program reviewed and approved by the NRC staff and performed at Davis-Besse Unit No. 1.

Installation of the redesigned SSHTs in the Davis-Besse Unit No. 1, Crystal River Unit No. 3 and TMI-2 reactor vessels did not present any unusual difficulties because it was performed prior to neutron activation of the reactor internals. Studies of methods to install the redesigned SSHTs in the irradiated B&W reactors, however, indicated that substantial difficulties would be experienced, primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to reinstall the SSHTs in TMI-1, they would cause significant radiation to personnel. Based on their experience in removing the SSHTs at TMI-1 and other reactors, B&W estimated that installing SSHTs in irradiated reactors would result in personnel exposure of about 100 man-rem per reactor. In the interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require reinstalling the SSHTs in TMI-1.

The proposed plan involves integrating the interrupted surveillance program at operating reactors which suffered damage to SSHTs into the programs for similar new plants. For TMI-1 this method for carrying out the program is in accord with the Site Integrated Surveillance Program permitted by Appendix H, 10 CFR 50 Paragraph II.C.4, in that the TMI-1 program would be conducted in TMI-2 and, therefore, would be conducted at the same site.

There are three distinct features of these proposed programs:

1. A host-reactor feature, in which the original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor which has been fitted with the newly-designed capsule holders;

2. An augmented surveillance feature in which more weld metal specimens and some larger fracture mechanics (compact tension (CT)) specimens will be included in the program; and

3. A data-sharing feature in which all available irradiation data for all of the beltline welds of a given reactor vessel will be considered by the licensee or his consultants in predicting the adjusted reference temperature and in making any fracture analyses for that vessel. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content, large and random variation of copper from point to point in the weld, and low initial upper shelf energy.

The specific program proposed for TMI-1 involves irradiating the remaining original TMI-1 surveillance capsules (one has been removed and tested) in some of the locations provided in the TMI-2 vessel. This plan will accomplish the original purpose of obtaining information on the effect of radiation on a material that is one of the controlling materials in the TMI-1 reactor vessel on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also will provide information from surveillance programs in Crystal River Unit No. 3, and Davis-Besse Unit No. 1 on material considered to be representative of the welds in the TMI-1 vessel. It is also important to note that still more information relevant to the TMI-1 vessel materials will be obtained from the NRC sponsored Heavy Section Steel Technology (HSST) irradiation programs. Details are provided below:

Three weld materials are of primary interest for the TMI-1 vessel. Procedure Qualification (P.Q.) numbers* WF 70 and WF 25 are used in the top and center

*Weld materials are specifically identified by the ASME Code by the procedure qualification test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

circumferential welds. The end of life (EOL) fluence for both of these welds is estimated to be 1.2×10^{19} nvt, and both have compositions that are expected to make them relatively sensitive to radiation damage. Weld P.Q. No. SA-1526, used for two of the longitudinal welds, also has high copper. Further, the EOL fluence at the azimuthal locations of these longitudinal welds is 9×10^{18} so they may become limiting during the service life. Another shell weld, the lower circumferential, is made of a material that is expected to be radiation sensitive (P.Q. No. WF 67), but the EOL fluence at this location is estimated to be at least an order of magnitude lower than that of the other circumferential welds, so it is not expected to be limiting.

The original TMI-1 surveillance material, WF 25, was the same as the center circumferential weld, so information on its radiation behavior is already available from the test results from one capsule irradiated in TMI-1. Two additional capsules containing this weld material will be irradiated in TMI-2 in accordance with the proposed program.

In addition to this TMI-1/TMI-2 integrated program, "research" capsules containing tensile, Charpy (C_V) and two sizes of compact tensile (CT) specimens of B&W archive materials will be included in the overall B&W power reactor surveillance program.

Weld materials included in the "research" capsules that may be limiting for the TMI-1 reactor vessel are: PQ SA-1526, PQ WF-25 and PQ WF-70. In addition, specimens of WF-67 material will also be included in the "research" program. These capsules will be irradiated in the Davis-Besse 1, Crystal River Unit No. 3 and TMI-2 reactor vessels. The presently planned withdrawal schedule calls for the first capsules to be withdrawn in 1981 to 1982, at which time the upper shelf energy of the specimens is predicted to be about 50 ft-lbs. Other "research" capsules will be withdrawn in about 1989 when the specimens will have received a fluence approximately equal to that at the inner surface of the vessel at EOL.

Also, research programs being sponsored by NRC will provide additional information on the effect of radiation on these specific weld materials and on several additional B&W weld materials expected to respond to radiation in a similar manner. These programs, HSST-2 and HSST-3, consist of many tensile, Cv, and CT specimens irradiated in a test reactor. Although information on the shift in RT_{NDT} will be obtained, the main emphasis of the HSST programs is to develop methods that can be used to better evaluate low shelf toughness using the rather small specimens used in the power reactor programs.

The following table shows where samples of the pertinent weld materials will be irradiated in the proposed integrated and "research" programs, what kinds of specimens will be used, and when the information will be available according to the present schedule.

Weld	Capsule Designation*	Irradiation Locations	Information Available	Specimen Types**
WF-70	R-1	Davis-Besse Unit No. 1	1981	Cv, CT
	R-2	Davis-Besse Unit No. 1	1989	Cv, CT
	R-1	Crystal River Unit No. 3	1982	Cv, CT
	R-2	Crystal River Unit No. 3	1989	Cv, CT
	HSST-3	Test Reactor	1978	Cv, CT
WF-25	TMI-1E	TMI-1	1976	Cv, Tensile (already removed)
	TMI-1A	TMI-2	1982	Cv, Tensile
	TMI-1C	TMI-2	1990	Cv, Tensile
	R-1	TMI-2	1982	Cv, CT
	R-2	TMI-2	1989	Cv, CT
	HSST-2	Test Reactor	1977	Cv, CT
	HSST-3	Test Reactor	1978	Cv, CT
SA-1526	R-1	TMI-2	1982	Cv, CT
	R-2	TMI-2	1989	Cv, CT
The following welds are not controlling for TMI-1:				
WF-67	R-1	Davis-Besse Unit No. 1	1981	Cv, CT
	R-2	Davis-Besse Unit No. 1	1989	Cv, CT
	R-1	Crystal River Unit No. 3	1982	Cv, CT
	R-2	Crystal River Unit No. 3	1989	Cv, CT
WF-8	-	-		
SA-1494	-	-		

Note: The irradiation schedule and withdrawal dates shown may be modified on the basis of initial test results to optimize the scheduled availability of information.

* & ** See top of page 6

* & ** from previous page

* R-1, R-2 - denotes "research" capsule
  TMI-1E - denotes capsule E from Three Mile Island Unit No. 1
  HSST-2, -3 - denotes capsule irradiated as part of the HSST program.
**Cv - denotes Charpy V-notch specimen.
  CT - denotes compact tensile specimen.

We have evaluated the effectiveness of this overall program plan, and have concluded that the information to be developed that is directly and indirectly relevant to the TMI-1 reactor vessel will be sufficient to provide assurance of safety margins against vessel failure that comply with Appendix G, 10 CFR 50.

Until data become available from the surveillance program, a conservative prediction of radiation damage can be made by using R.G. 1.99* for at least the next five years of operation. This Regulatory Guide is based on the NRC staff's analysis of all data available at the time it was written. New data, in particular the results of the augmented integrated surveillance program described above, will be used to periodically update the Regulatory Guide. Predictions of the adjustment of reference temperature and the drop in upper shelf energy are given graphically in R. G. 1.99 as functions of copper and phosphorus content and of fluence. In addition there is an "Upper Limit" line on each graph, which is to be used when information about the copper and phosphorus contents is inadequate. Because the chemical analyses of the B&W welds have shown considerable variation, the NRC staff intends to use the Upper Limit lines as the basis for any predictions required at this time.

We have also considered the uncertainties involved in applying radiation effects information obtained in other reactors to the TMI-1 vessel. The major uncertainties involved are:

1. Accuracy of neutron fluence calculations;

2. Magnitude and effect of variation in neutron spectra between reactors;

3. Magnitude and effect of variations in irradiation temperature between reactors;

4. Magnitude and effect of variations in rate of irradiation on material properties.

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*Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", July 1975. Revision 1 is to be published in April 1977.

The effects of these variables have been studied for at least 20 years. Although some uncertainties still remain, the effects are fairly well established and understood as discussed below.

1. Calculational methods for estimating the neutron flux at the reactor vessel wall and at irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. As a result, the fast neutron flux and fluence in these locations can generally be calculated to an accuracy of + 20%, particularly if some dosimetry checks are available. Dosimeters from the original TMI-1 surveillance program were removed and tested, so the fluence calculations for the vessel have been verified.

   In addition, it should be noted that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence; hence, uncertainties of 20 to 50% in fluence are not highly significant.

   We have also considered the fact that the design of the TMI-1 vessel, internals, and core is nominally identical to that of the other reactors which will be used to obtain radiation effects information.

   These considerations are the basis for our conclusion that uncertainties in the calculation of neutron fluence will be small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when this is evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W reactor to another are stated to be relatively minor, because they have similar geometry.

   We considered the possible differences in neutron spectra that could occur between the B&W power reactors involved in the integrated program. Such effects can be dealt with, if necessary, through methods that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects. The neutron spectrum uncertainty will be kept under active scrutiny by the NRC staff and if additional developments (theoretical or experimental) suggest that the effect might be significant under some conditions, appropriate adjustments in reference temperature, drop in upper shelf energy or other suitable parameter can be made.

3. The effect of the temperature of irradiation has also been the subject of considerable reserach. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into account if the irradiation temperature difference is over about 25°F. Sufficient information

is available to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and and vessel walls between the B&W power reactors involved in the integrated program are expected to be less than 50°F, and can be conservatively evaluated.

4. The effect of irradiation rate has been evaluated by research programs at the Naval Research Laboratory (NRL) and other laboratories. Although the consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude, more data from surveillance programs are needed to provide verification. However, the differences in the rates of irradiation of specimens in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude. Therefore, we have concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

To implement the proposed program for TMI-1, the licensee has proposed Technical Specifications which would require the remaining TMI-1 capsules to be irradiated in TMI-2 on a schedule consistent with the overall integrated and augmented surveillance program. We have found that certain revisions should be made in the proposed Technical Specifications. These revisions have been discussed with and found acceptable by the licensee. In addition, to assure that timely accumulation of neutron exposure by the specimens is not unduly affected by extended outages at TMI-2, we have suggested to the licensee, additional Technical Specifications which would require the proposal of alternate irradiation plans if irradiation of the specimens was not being accomplished in a timely manner at TMI-2. The licensee has agreed to these suggested additional Technical Specifications.

Finally, it is noted that the proposed overall integrated, augmented program (with possible minor modification yet to be finalized) should provide more useful information than could have been extracted from the original surveillance program. The proposed program will also provide information of the type needed to meet the requirements of Paragraph V.C of Appendix G, 10 CFR 50.

Conclusion Regarding Reactor Vessel Material Surveillance Program

We have evaluated the adequacy of the proposed integrated, augmented reactor vessel material irradiation program for TMI-1 as an alternative to the original program that was interrupted by failure of the associated hardware. We conclude that the proposed program will provide the information required to comply with Appendix G, 10 CFR 50, and that the degree of commonality between TMI-1 and TMI-2 and the predicted severity of irradiation is such that the uncertainties involved in using data obtained from surveillance specimens irradiated in Three Mile Island Unit 2 to establish TMI-1 vessel operating limitations are small and can be accounted for

by imposition of appropriate margins. We also conclude that the associated Technical Specification changes to implement the program are acceptable.

2. Fuel Reload for Cycle 3 Operation

Reload Description

The TMI-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The reloading for Cycle 3 operation (2, 3) consists of the removal of all batch 2 assemblies, the relocation of batch 3 and 4 assemblies, and the introduction of 13 batch 1a and 48 new batch 5 assemblies. The batch 5 assemblies will be located at the core periphery and the batch 1a assemblies will occupy 13 positions within the mixed central zone.

Fuel Mechanical Design

The outside dimensions and configuration of the Mark B-4 (batch 4 & 5) fuel assemblies and older Mark B-3 (batch 3) fuel assemblies are identical except that the Mark B-4 have spring-type flexible spacers and the Mark B-3 have corrugated-type flexible spacers. This Mark B-4 fuel rod spacer has been previously reviewed and found acceptable by the NRC staff on the basis of no significant mechanical or material change to the reactor operation (4) and has been successfully operating in similar cores for a substantial time (Section 4.5 of Reference 1). The Mark B-4 fuel assemblies, therefore, do not represent any unreviewed or untested change in mechanical design from the reference cycle and are therefore acceptable.

This mechanical design change has been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that this fuel design difference in the TMI-1 core is of negligible effect.

Fuel rod cladding creep collapse analyses were performed for the Cycle 3 core. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse. (1, 5) The calculational methods, assumptions, and data have been previously reviewed and approved by the NRC staff (6). The analysis assumed a 2000 hour densification time which maximizes creep; no fission gas production which maximizes differential pressure; and a lower tolerance limit on clad thickness and an upper tolerance limit on cladding ovality, both of which maximize cladding creep deformation.

The batch 3 fuel was found to be more limiting than the batch 4, 5, and 1a fuel due to the lower prepressurization, lower pellet density, and previous power history. The most limiting assembly in batch 3 was found to have a collapse time longer than the maximum projected three-cycle core exposure (24,288 EFPH).

From the viewpoint of cladding stress due to differential pressure, thermal stress due to fuel temperature gradients, and bending stress, neither the yield stress nor the B&W 1% total strain criterion for the cladding is predicted to be exceeded in the Cycle 3 core.

The batch 5 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previously successful operations with equivalent fuel, we conclude that the fuel mechanical design for Cycle 3 operation is acceptable and its application to Cycle 3 operation will not endanger the health and safety of the public.

Fuel Thermal Design

The fuel thermal design analysis was conducted with the TAFY-3 computer code, as discussed in reference 7. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in the "Fuel Densification Report".(8) Modifications to the "Fuel Densification Report" on the fuel pellet void probability, F_q, and fuel grain size distribution, F_k have been previously reviewed and approved by the NRC staff. (9)

Based on the analyses presented in reference 1 and comparison with allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations, (12) the fuel thermal design for the TMI-1 Cycle 3 core is acceptable and can be applied with reasonable assurance that the health and safety of the public will not be endangered.

Fuel Material Design

The fuel material design for Cycles 2 and 3 operation is not significantly different from that of Cycle 1 operation. The only difference is that Zircaloy-4 is used as the fuel assembly tubular spacer material in Mark B-4 fuel instated of zirconium dioxide (ZrO_2), which is used in Mark B-3 fuel. This change does not affect the fuel system chemistry. This change has been reviewed and has a substantial amount of previous experience (Section 4.5 of reference 1). Therefore, the fuel material design for TMI-1 Cycle 3 operation is acceptable.

Nuclear Design

The TMI-1 reactor has completed two operating cycles and is thus sufficiently close to equilibrium cycle to show only minor changes in physics parameters. The Cycle 3 core will consist of four distinct fuel types: fresh batch 5 assemblies located at periphery, once-burned batch 4 assemblies located generally in

an intermediate zone and also near the core center, twice-burned batch 3 assemblies located between the periphery and the intermediate zone, and located between the intermediate zone and central zone, plus 13 batch 1a assemblies loaded with the batch 4 assemblies. Thus, although the Cycle 3 core is a four batch loading, the physics parameters are quite close to those of the Cycle 2 core. In addition, these parameters will be verified during the startup testing program described later.

The only significant procedural change from the reference cycle (Cycle 2) is the specification of axial power shaping rod (APSR) position limits. The APSR position limits will provide additional control of power peaking through an improved definition of the core power distribution.

The calculational methods used by the licensee are the same as were used for cycle 2.(10) Because of this, and because of the verification provided by the physics testing which will be performed during the Cycle 3 startup, we find the nuclear design for Cycle 3 to be acceptable.

Thermal-Hydraulic Analysis

Major acceptance criteria for the thermal-hydraulic design are specified in the NRC's Standard Review Plan Section 4.4 ("Thermal and Hydraulic Design"). These criteria establish the acceptable limits for DNBR (Departure from Nucleate Boiling Ratio). The thermal-hydraulic analyses for the TMI-1 Cycle 3 reload core were made with previously approved models and methods, as stated in the TMI-1 Final Safety Analysis Report.(11)

The reactor coolant flow rate was accurately measured during Cycle 1 operation and a minimum measured value of 108% of the system design flow was determined. The licensee has taken credit in Cycles 2 and 3 thermal-hydraulic analyses for the fact that the actual system flow is greater than the design flow rate, and has also included uncertainties and conservatisms in this analysis.(1, 10) The new design flow is 106.5% of the Cycle 1 design flow.

In the past, a reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for TMI-1. This penalty was associated with the potential for a core internal vent valve to be stuck open during normal operation. The core internal vent valves are incorporated into the design of the reactor internals to preclude potential vapor lock during a postulated cold-leg break Loss-of-Coolant Accident (LOCA). We have concluded that by application of a surveillance program the vent valve flow penalty may be removed. The surveillance requirements demonstrate that the vent valves are not stuck open and that the vent valves operate freely. A separate review of the licensee's surveillance program for the vent valves has concluded that the program adequately meets our requirements, and that the vent valve penalty was properly eliminated (12).

The effect of fuel rod bow was evaluated by the licensee with consideration given to both the hot channel power spike and the effect on DNBR. This evaluation was also separately reviewed and accepted by the NRC staff (12).

There are differences in the flow resistance between the Mark B-3 fuel assemblies and the Mark B-4 assemblies. The flow resistance for a Mark B-4 fuel assembly is slightly less than that for the Mark B-3 assemblies. For the Cycle 3 loading, the highest assembly power always occurs in a Mark B-4 assembly. The Cycle 2 analysis(10), also used for Cycle 3 reference evaluation(1), assumed the hot assembly to be a Mark B-3 type. This analysis is conservative for Cycle 3 because the predicted hot assembly coolant flow rate is less than that of a corresponding Mark B-4 assembly.

Because of the analyses discussed above, we have found the thermal-hydraulic analysis to be acceptable and the proposed Technical Specifications related to the thermal-hydraulic analysis also acceptable.

Accident and Transient Analyses

A generic LOCA analysis for a B&W 177 assembly lowered-loop plant has been performed using the Final Acceptance Criteria ECCS evaluation model (13, 3). This analysis has been reviewed by the NRC staff (14), and found applicable to the TMI-1 Cycle 3 core.

All other accidents and transients (loss of flow, dropped rod, inadvertent bank withdrawal, etc) have been examined by the licensee for Cycle 3 and found to fall within the bounds of the FSAR analyses, as updated for Cycle 2 operation. We have reviewed the various input parameters for Cycle 3, and have found the licensee's conclusion acceptable.

Startup Program

The licensee has proposed a startup program which will verify:

Critical boron concentration.

Temperature reactivity coefficient at two points.

Control bank worth by boron swap. More than half the required shutdown reactivity will be verified using this method.

Control bank worth by bank drop. The remainder of the banks will be checked by this method.

Ejected rod worth.

In addition, during the power escalation phase, the startup program will verify:

Power distribution at three plateaus.

Dropped-rod power distribution.

Incore/excore imbalance correlation.

Doppler coefficient at 100% power.

Moderator temperature coefficient at 100% power.

We have reviewed this proposed startup program and have found it acceptable.

Technical Specifications

The licensee has proposed revisions to the Technical Specifications to implement the changes due to the Cycle 3 reload(1). We have reviewed the revised Technical Specifications relating to Cycle 3 operation and found them acceptable except in two instances: (1) the Technical Specifications did not define an acceptable range of values for the moderator temperature coefficient below 95% power, and (2) the intervals at which core power maps would be obtained required clarification. These deficiencies were brought to the attention of the licensee who proposed revised Technical Specifications (3) which we find acceptable.

Conclusion Regarding Fuel Reload for Cycle 3 Operation

Based on our evaluation of the application as set forth above, we conclude that the proposed changes in Technical Specifications associated with operation in Cycle 3, as revised, are acceptable.

Based on the foregoing evaluation of fuel and nuclear design, thermal-hydraulic performance and accident analyses, we also conclude that the operational characteristics of TMI-1 in Cycle 3 will be substantially the same as those in Cycle 2 and that operation of TMI-1 as proposed, will not involve a significant increase in the probability or consequences of accidents previously considered nor involve a significant decrease in safety margin. Accordingly, we have concluded that operation of TMI-1 does not involve a significant hazards consideration.

3. Pressure Limits During Reactor Coolant System Heatup and Cooldown

By letter dated February 23, 1977, the licensee proposed reactor coolant system (RCS) pressure-temperature operating limits to be applicable through six effective full power years (EFPY) of operation. The calculated maximum fluence on the vessel wall interior surface at the end of 6 EFPY was stated to be 3.2×10^{18} n/cm^2.

There are six different weld materials in the reactor vessel beltline region. These are designated WF-8, WF-25, WF-67, WF-70, SA-1494 and SA-1526. Based on the chemical compositions and locations of these welds in the beltline, WF-25, WF-70 and SA-1526 are expected to be the most limiting materials.

Weld material WF-25 is included in the TMI-1 material surveillance program. It contains approximately 0.34% copper and 0.015% phosphorus, and its initial reference temperature (RT_{NDT}) was -14°F. Capsule TMI-1E, containing this weld material, was removed from TMI-1 following the first operating cycle. The dosimeters installed in this capsule indicated that the specimens had received a fast fluence of approximately 1.07×10^{18} n/cm^2 during their exposure. Examination of the WF-25 specimens which had received this fluence indicated that the upper shelf energy of the material had decreased from 81 ft-lbs to 62 ft-lbs, and that the shift in RT_{NDT}, measured at 50 ft-lbs, was 117°F. Both of these experimental values agree closely with the values predicted by Regulatory Guide 1.99.

The other possibly limiting weld materials are WF-70 and SA-1526. WF-70 contains approximately 0.27% copper and 0.014% phosphorus and an estimated initial RT_{NDT} of 20°F. SA-1526 contains approximately 0.36% copper and 0.016% phosphorus and also has an estimated initial RT_{NDT} of 20°F.

Based on the composition, initial RT_{NDT} and neutron fluence at the locations of the various beltline materials, the licensee estimated, using the curves in Regulatory Guide 1.99, Revision 1*, the adjusted RT_{NDT} at 6 EFPY for each of these materials. From these adjusted values of RT_{NDT}, the licensee then developed the proposed operating limit curves. The most limiting material according to the licensee's analysis was SA-1526 which had an adjusted RT_{NDT} of 145°F at the end of 6 EFPY.

We have reviewed the licensee's submittal and have determined that the adjusted RT_{NDT} for material WF-70 was estimated by the licensee based on the measured copper content of this material. However, in view of the variances we have found in the chemical composition of some welds (particularly copper content), we believe the upper limit lines of Regulatory Guide 1.99 should be used to predict changes in RT_{NDT}. When this is done for the WF-70 material, it is found to have an adjusted RT_{NDT} of 160°F at 6 EFPY and, thus, is more limiting than the SA-1526 used by the licensee.

At 5 EFPY, however, WF-70 would have an adjusted RT_{NDT} of less than 145°F. Therefore, since the proposed operating limit curves were based on a maximum adjusted RT_{NDT} of 145°F, reduction of the term of applicability of the curves from 6 EFPY to 5 EFPY would allow the curves to conservatively represent all of the limiting materials - including WF-70. Such a reduction has been discussed with and found acceptable by the licensee.

* To be published April, 1977.

Conclusion Regarding Revised Pressure-Temperature Limit for Heatup and Cooldown

Based on the above considerations, we conclude that the proposed operating limit curves, limited in applicability to 5 EFPY, are acceptable and conform to the requirements of 10 CFR 50, Appendix G,

No changes were proposed for Technical Specification 3.1.3, "Minimum Conditions for Criticality". This specification requires that the reactor coolant temperature be above 525°F prior to criticality except for low power physics tests. This specification meets the requirements of paragraph 4.A.2.c of Appendix G, 10 CFR Part 50 and is acceptable.

Conformance with Appendix G, 10 CFR Part 50 in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements to NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Because this proposed change is in conformance with Appendix G to 10 CFR 50 which is designed to maintain an adequate safety margin, we conclude that the proposed change, as revised, does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in safety margin. We therefore, further conclude that the proposed change, as revised, does not involve a significant hazards consideration.

Environmental Consideration

We 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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusions

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.

Dated: April 22, 1977

References

1. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated January 26, 1977, enclosing Technical Specification Change Request No. 45 and BAW-1442.

2. BAW-1442, Three Mile Island Unit 1 Cycle 3 Reload Report, November, 1976.

3. Letter, R. C. Arnold (Metropolitan Edison) to Director of Nuclear Reactor Regulation, dated March 31, 1977, enclosing responses to Round 1 Questions concerning the TMI-1 Cycle 3 Reload Application.

4. SE on Oconee Nuclear Station, Units 1,2,&3, dated June 30, 1976, Amendment Nos. 27, 27, and 23 for License Nos. DPR-38, DPR-47, and DPR-55.

5. BAW-10084P, Rev. 1, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, October, 1976

6. Letter, A. Schwencer (NRC) to J. F. Mallay (B&W), dated January 29, 1975

7. BAW-10044, TAFY - Fuel Pin Temperature and Glass Pressure Analysis, May, 1972

8. BAW-10055, Rev. 1, Fuel Densification Report, June, 1973

9. Memorandum from R. Lobel to D. F. Ross, "Present Status of B&W Power Spike Model," July 23, 1974.

10. Three Mile Island Unit 1 - Cycle 2 Reload Report, Revision 1, July 1, 1976.

11. Three Mile Island Unit 1 Nuclear Station, Final Safety Analysis Report, USNRC Docket No. 50-289.

12. Safety Evaluation by the Office of Nuclear Reactor Regulation, Amendment No. 25 to Facility Operating License No. DPR-50, dated March 7, 1977.

13. BAW-10103, Rev. 1, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, September, 1975.

14. Letter, D. F. Ross (NRC) to D. B. Vassallo (NRC), Re: Topical Report Evaluation BAW-10104, ECCS Evaluation Model, Revised Nucleate Boiling Lockout Model, dated February 2, 1977.