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AEP:NRC:1067
10 CFR 50.90

Donald C. Cook Nuclear Plant Unit 1
License No. DPR-58
Docket No. 50-315
REDUCED TEMPERATURE AND PRESSURE PROGRAM
ANALYSES AND TECHNICAL SPECIFICATION CHANGES

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

October 14, 1988

Dear Dr. Murley:

This letter and its attachments constitute an application for amendment to the Technical Specifications (T/Ss) for the Donald C. Cook Nuclear Plant Unit 1. Specifically, these changes are those required to support the upcoming Unit 1 Cycle 11 fuel cycle. This cycle will incorporate operation of the unit at reduced temperatures and pressures with the intent of reducing steam generator U-tube stress corrosion cracking of the kind observed in Cook Nuclear Plant Unit 2. These changes are necessary prior to the unit entering Mode 3 after its refueling. We presently anticipate Mode 3 to occur as early as April 22, 1989. Therefore, we request that your review of the changes be completed by April 15, 1989.

The T/S changes are supported by the attachments to this letter. Attachment 1 contains a description of the changes and our evaluation concerning significant hazards considerations. Attachment 2 discusses analyses and evaluations performed by Westinghouse Electric Corp. (Westinghouse) that address the impact of the Reduced Temperature and Pressure (RTP) Program on the accident analyses and on primary and auxiliary components and systems. The Westinghouse report is contained in a separate binder and accompanies this letter. Appendix A to the Westinghouse report contains marked-up FSAR pages to accompany the LOCA analyses found in the report. Attachment 3 to this letter provides evaluations of the effect of the RTP program on balance of plant systems, as well as miscellaneous safety

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analyses that were outside the scope of the Westinghouse report. Attachment 4 contains the proposed revised T/S pages. (This attachment is referred to as Appendix B in the Westinghouse report.) Several of the revised T/S pages provided in this submittal are affected by previous requests that are pending at the NRC. The affected pages and submittals are as follows:

Submittal No.	Date	Page
AEP:NRC:0692AJ	May 30, 1986	3/4 7-5
AEP:NRC:0916W	March 26, 1987	1-7, 2-8, 2-9, 3/4 2-5, 2-7, 2-8, 2-9, 2-14, 2-15, 7-5, B 2-1 (a), B 2-5, B 3/4 2-1
AEP:NRC:1072	September 15, 1988	3/4 5-5

With the exception of the change to page 3/4 7-5 proposed in our letter AEP:NRC:0916W, the changes proposed in the present submittal are in addition to the previous changes and do not supersede them. (The previous changes to page 3/4 7-5 are discussed in detail in Attachment 1.) During discussions on September 30, 1988, your staff indicated that the changes proposed in AEP:NRC:0916W were expected to be granted in the near future. Therefore, we have included those pending changes in the proposed revised T/S pages submitted with this letter.

This letter is the second of two transmitting analyses related to the Unit 1 RTP Program. An analysis of containment long-term pressure following a LOCA was submitted via our letter AEP:NRC:1024D, on August 22, 1988.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amounts of any effluent that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed changes have been reviewed by the Plant Nuclear Safety Review Committee and will be reviewed by the Nuclear Safety and Design Review Committee at their next regularly scheduled meeting.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to Mr. R. C. Callen of the Michigan Public Service Commission and Mr. George Bruchmann of the Michigan Department of Public Health.

Per the requirements of 10 CFR 170.12(c), we have enclosed an application fee of $150.00 for the proposed amendments.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

MPA/dat

Attachments

cc: D. H. Williams, Jr.
W. G. Smith, Jr. - Bridgman
G. Bruchmann
R. C. Callen
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman

ATTACHMENT 1 TO AEP:NRC:1067

DESCRIPTION AND 10 CFR 50.92 SIGNIFICANT HAZARDS EVALUATION

FOR CHANGES TO THE TECHNICAL SPECIFICATIONS

FOR THE DONALD C. COOK NUCLEAR PLANT UNIT 1

Introduction

The technical specification (T/S) changes proposed in this letter support operation of Unit 1 of the Donald C. Cook Nuclear Plant at reduced temperature and pressure (RTP) conditions. The program was undertaken to prevent stress corrosion cracking of steam generator tubes of the type experienced in Unit 2 of the Cook Nuclear Plant. Westinghouse Electric Corp. (Westinghouse) has determined that the reduction in pressures and temperatures should more than double the time elapsed to reach a given level of corrosion with respect to the original design temperatures and pressures.

Technical Specification Revisions

The T/S changes found in Attachment 4 are based on the RTP analyses performed by Westinghouse. The changes provide consistency between the T/Ss and the analyses, or clarify the T/Ss with regard to the RTP analyses. The changes are described below.

Page 1-7

Definition 1.38 (design thermal power) is being deleted, and the present Definition 1.39 (allowable power level) renumbered accordingly. The majority of the RTP analyses performed by Westinghouse assumed a core power level of 3413 Mwt. This core power level is 2 Mwt higher than the value presently listed for design thermal power. Rather than revise the definition, however, we have chosen to delete it. The reactor core is only licensed to operate up to 3250 Mwt, which is the rated thermal power defined by Definition 1.3. The accident analyses are done at a higher power level in order to position the unit for a potential power uprating. The definition of design thermal power does not reflect our licensed power level and therefore adds an unnecessary complication to the T/Ss. In addition, not all of the analyses are performed at the design thermal power level. For example, the small break LOCA analysis was performed at a core power level of 3588 Mwt in order to bound potential power uprating for both of the Cook Nuclear Plant units. The large break LOCA with residual heat removal (RHR) cross-tie valves closed and some other evaluations were performed at 3250 Mwt. Thus, there is no single value for design thermal power that would correspond to all the analyses. For these reasons, we believe the definition should be deleted. The power level assumed for each of the accident analyses will be clearly identified in the updated FSAR, which is the proper location for this type of information.

Page 1-10

Table 1-3 has been deleted. This table provides information on which analyses were performed at the design thermal power level. Deletion of the table is consistent with deletion of the term design thermal power as discussed above.

Page 2-2

Figure 2.1-1 (core safety limits) is being revised. The revision is based on the revised DNBR safety limit value of 1.45, which is discussed in Section 3.3.2.1 of the Westinghouse report. The revised core safety limits are accompanied by changes to the overtemperature and overpower delta T trip setpoints, as described below.

Page 2-5

The pressurizer pressure low setpoint is conservatively increased by 10 psig. The revised T/S setpoint is based on the setpoint for the function assumed in the large and small break LOCA analyses. See Section 3.14 of the Westinghouse report.

Pages 2-7, 2-8

The T/S overtemperature delta T trip setpoint equation is being revised based on the setpoint equation used by Westinghouse in the non-LOCA DNB transients. The equation is conservatively redefined in terms of rated thermal power rather than design thermal power, consistent with the elimination of design thermal power as a defined term, as discussed above. The revised setpoint equation protects the revised core safety limits discussed previously. Illustration of the overtemperature delta T setpoints relative to the core safety limits is provided in Figures 3.3-1a through 3.3-1d of the Westinghouse report. The nominal temperature, T' has been limited to 567.8^{o}F. This is consistent with the maximum T_{avg} supported by the complete spectrum of the RTP evaluations, as discussed in Sections 2.1 and 3.3.4 of the Westinghouse report. T' and P' will be set at the actual nominal operating values of their respective plant parameters to ensure that trips occur at approximately the analyzed time as discussed in Section 3.3.2.1 of the Westinghouse report and in the proposed bases Section 2.2.1 for the overtemperature delta T trip. Nominal pressure, P', has been specified as either 2235 psig or 2085 psig. These values are consistent with the two allowable pressures supported by the Westinghouse analysis, as discussed in Section 3.3.2.1 of the Westinghouse report. The complete equation for the overtemperature delta T setpoint used by Westinghouse is provided in Section 3.14 of the Westinghouse report.

Page 2-9

The overpower delta T trip setpoint equation is being revised. Although not taken credit for in the Westinghouse analysis, the overpower delta T setpoint had to be revised to reflect the revised core safety limits discussed previously. Illustration of the overpower delta T setpoint relative to the core safety limits is provided in Figures 3.3-1a through 3.3-1d of the Westinghouse report. The nominal average temperature, T'', has been limited to 567.8^{o}F. This is consistent with the maximum T_{avg} supported by the complete spectrum of the RTP evaluations, as discussed in Sections 2.1 and 3.3.4 of the Westinghouse report. T'' will be set at the actual nominal operating temperature of the plant as was the case for T' discussed above. The complete equation for the overpower delta T setpoint equation used by Westinghouse is provided in Section 3.14 of the Westinghouse report.

Page 3/4 2-5

The maximum value of $F_Q(Z)$ is increased from 2.10 to 2.15. The increased value is supported by the large break LOCA analysis performed by Westinghouse (See Sections 3.1.1 and 3.14 of the Westinghouse report.)

In addition, the F_Q limits applicable to Exxon Nuclear Co. (Exxon) fuel are being deleted. The fuel supplier for Unit 1 was switched in Cycle 8 from Exxon to Westinghouse. At the present time, no Exxon assemblies are in use or scheduled for reinsertion. The analyses performed by Westinghouse for the RTP program do not support use of Exxon fuel at reduced temperature or pressure. Therefore, we are deleting the reference to limits on the Exxon fuel. This change has the added benefit of simplifying the T/Ss by removing information that cannot be used.

Page 3/4 2-7

The K(Z) curve applicable to Exxon fuel is being deleted.

Page 3/4 2-8

The K(Z) curve applicable to Westinghouse fuel is being revised to reflect the new LOCA analysis performed by Westinghouse. (See Section 3.14 of the Westinghouse report.)

Page 3/4 2-9

The $F^N_{\Delta H}$ limit applicable to Exxon fuel is being deleted.

Page 3/4 2-14

The DNB table is being revised to reflect the RTP analyses, as discussed below.

Reactor Coolant System T_{avg}

The present upper limit on T_{avg} is increased slightly to 570.9°F. This limit is consistent with the upper temperature limit supported by the complete spectrum of RTP analyses. The controller deadband and measurement error allowance was increased from the previous analysis value of 4.0 to 4.5°F, resulting in a 0.5°F increase in allowable T_{avg}. (See Sections 3.3.3.1 and 3.14 of the Westinghouse report.)

Pressurizer Pressure

The present lower limit on pressurizer pressure is decreased to 2050 psig. This limit is consistent with the pressure value supported by the RTP analyses for operation at 2085 psig including allowance for uncertainty. The uncertainty in the steady state fluctuations and measurement error allowance was increased to 35 psi from the previous analysis value of 30 psi. The value of 2050 psig was obtained by adjusting the nominal pressure of 2085 psig for uncertainties. (See Sections 3.3.3.1 and 3.14 of the Westinghouse report.)

Reactor Coolant System Total Flow Rate

The reactor coolant system minimum measured total flow (MMF) rate is changed from 138.6 x 10^6 lb/hr to 366,400 gpm. This volumetric flow rate is consistent with the analysis value for flow of 354,000 gpm assumed in the non-DNB transients. The analysis value is 3.5% less than the MMF. The MMF is used directly in the DNB analyses performed using the Westinghouse Improved Thermal Design Procedures. The Westinghouse analyses assume a constant volumetric flow over the complete range of temperatures and pressures supported by the RTP analyses. The total mass flow will be determined under test conditions. The mass flow will be converted to volumetric flow entering the core using the cold leg temperatures from the test data and compared to the MMF during surveillance of primary total flow. (See Section 3.14 of the Westinghouse report.)

Page 3/4 2-15

The F_Q limit in the definition of APL is changed to 2.15 for Westinghouse fuel. This change is consistent with the change

proposed for the F_Q limit of Specification 3.2.2 described above. Limits on APL applicable to Exxon fuel are deleted.

Page 3/4 3-10

Functional Unit 2

The power range neutron flux reactor trip functional unit is being clarified by indicating that the 0.5 second response time is applicable to both the low and high setpoints of the function. This change is editorial in nature.

Functional Unit 11

The pressurizer water level-high reactor trip response time is being changed from "not applicable" to 2.0 seconds. The trip was modeled in the analysis of the uncontrolled rod withdrawal at power event. The trip precludes filling of the pressurizer for that event. (See Section 3.3.4.4 and Table 3.3-2 of the Westinghouse report.)

Pages 3/4 3-24, 26

Functional Unit 1.f and 4.d

The steam line pressure-low setpoint is decreased by 100 psig in functional units 1.f and 4.d. The new nominal setpoint for both functional units is 500 psig with a 480 psig allowable value. The changes to functional units 1.f and 4.d are supported by the steam line break analysis discussed in Section 3.3.4.13 and the steam line mass and energy release evaluations described in Section 3.3.4.1. of the Westinghouse report. The allowable value is provided in Section 3.14 of the Westinghouse report.

Page 3/4 4-6

The maximum pressurizer level is being increased to 92% of span. The purpose of the maximum pressurizer level limit, as described in the Bases, is to ensure that a bubble can exist in the pressurizer. Westinghouse has determined that a bubble can be maintained at the 92% level. The change is described in detail in Section 3.13 of the Westinghouse report. The change will allow operational flexibility at the higher end of the T_{avg} spectrum analyzed for the RTP program.

Page 3/4 5-1

The lower limit on required accumulator water volume is reduced from 929 to 921 ft^3. The change is consistent with the minimum

accumulator volume supported by the Westinghouse small and large break LOCA analyses. (See Section 3.14 of the Westinghouse report.)

Page 3/4 5-5

The proposed change reduces the required value of discharge pressure for surveillance tests for the safety injection pumps and the RHR pumps. These changes are justified by analyses performed as part of the RTP Program. The LOCA analyses are discussed in Section 3.1 of the Westinghouse report. The containment LOCA mass and energy release is described in WCAP-11908 entitled, "Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2." This WCAP, which is part of the RTP Program, was submitted on August 22, 1988, attached to our letter AEP:NRC:1024D.

We are not requesting a change in the centrifugal charging pump discharge pressure at this time because the steam line break mass/energy releases for inside and outside containment were evaluated, but not reanalyzed for this submission. These analyses are discussed in Section 3.3.4 of the Westinghouse report. (See also Section 3.14 of the Westinghouse report.)

Page 3/4 5-6

A requirement is being added to verify that the charging pump discharge resistance is within a specified range following modifications to the ECCS that alter the flow characteristics. The allowable range of discharge resistances ensures that the flow to the core delivered by the charging pumps in the event of a small break LOCA is within the analyzed values. (See Section 3.14 of the Westinghouse report.) The requirements of the footnote on page 3/4 5-6 are broken into four parts, a through d, for clarity.

Page 3/4 7-5

The discharge pressure requirements for the motor- and turbine-driven auxiliary feedwater pumps are established as 1375 psig and 1285 psig, respectively. This is the value currently in the T/Ss. Our letter AEP:NRC:0916W, dated March 26, 1987, proposed to decrease the required discharge pressures to 1240 psig for the motor driven pump and 1180 psig for the turbine driven pump.

The feedwater line break analysis is not in the licensing basis for Unit 1 of the Cook Nuclear Plant. However, an evaluation of the accident was performed and supported 15% degradation of the pumps from the manufacturer's pump head curve. (The present T/S discharge pressures correspond to 5% degradation.) The conditions of the RTP program were not addressed in the

evaluation supporting the AEP:NRC:0916W changes. An analysis performed in conjuction with the RTP Program showed that acceptable results could be obtained under the assumptions of the RTP Program if 5% degradation was assumed. Therefore, the AEP:NRC:0916W proposed values are being replaced with the present T/S values.

Page B 2-1(a)

The design limit DNBR and safety analysis limit DNBR are being revised to the limits for the RTP Program. The DNBR limits are discussed in Section 3.3.2.1 of the Westinghouse report.

Additionally, DNB limits applicable to Exxon fuel are being deleted.

Page B 2-2

Limits on $F^N_{\Delta H}$ applicable to Exxon fuel are being deleted and reference to design thermal power is being deleted.

Page B 2-4

The Bases section pertaining to the overtemperature delta T trip function is being revised to reflect the limits on reference average temperature and reference operating pressure that were discussed above.

Page B 2-5

Overpower delta T

The Bases section pertaining to the overpower delta T trip function is being revised to reflect the changes to the function described above.

Pressurizer Water Level

The Bases description of the pressurizer water level trip is being revised to indicate that the trip is necessary to preclude filling of the pressurizer in the event of an uncontrolled RCCA withdrawal at power event. This is discussed in Section 3.3.4.4 and Table 3.3-2 of the Westinghouse report.

Page B 3/4 2-1

The minimum DNBR value of 1.69 in the introduction to the Bases for Section 3/4 2 is replaced with the term "the safety limit DNBR." The safety limit DNBR supported by the RTP program is being lowered to 1.45. This is addressed by our proposed change to page B 2-1(a). Use of the term "safety limit DNBR" will avoid

having to revise Page B 3/4 2-1 again should future analyses modify the safety analysis limit DNBR.

Page B 3/4 6-2

The maximum calculated containment pressure resulting from a LOCA is being revised to 11.89 psig. This value is consistent with that reported in WCAP 11908, entitled "Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2." This WCAP, which is part of the RTP Program, was submitted on August 22, 1988, via our letter AEP:NRC:1024D.

10 CFR 50.92 Evaluation

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

1) Involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

3) Involve a significant reduction in a margin of safety.

Criterion 1

The T/S changes required by the RTP program are accompanied by extensive analyses and evaluations which indicate that the program will not result in an unsafe condition at the plant. The analyses and evaluations support our conclusion that the changes will not involve a significant increase in the probability or consequences of a previously analyzed accident, nor will they involve a significant reduction in a margin of safety.

Criterion 2

The RTP analyses and evaluations have been demonstrated to comply with the licensing basis of the plant. Thus, the program should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

See Criterion 1 above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to

involve significant hazards consideration. The sixth of these examples refers to changes that may result in some increase to the probability or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable.

The RTP analyses have been demonstrated to comply with the licensing basis of the plant. Thus, we believe the example cited is applicable and that the changes should not involve significant hazards consideration.

ATTACHMENT 2 TO AEP:NRC:1067

RESULTS OF WESTINGHOUSE ANALYSES

PERFORMED IN SUPPORT OF THE

REDUCED TEMPERATURE AND PRESSURE PROGRAM

Description of Attachment 2

Attachment 2 contains the results of analyses performed by Westinghouse Electric Corp. (Westinghouse) in support of the Reduced Temperature and Pressure (RTP) Program, Areas within the Westinghouse scope included reviewing the FSAR Chapter 14 accidents, containment analyses and primary and auxiliary components and systems. Appendix A to the attachment contains marked-up pages from the Cook Nuclear Plant Updated FSAR. The marked-up pages accompany the small and large break LOCA analyses discussed in Section 3.1 of the Westinghouse report. Appendix B of the Westinghouse report contains the proposed revised T/S pages. These changes are described in detail in Attachment 1.

The RTP Program for Unit 1 supports a continuous range of full power vessel Tavg between 547.0°F and 567.8°F, and reactor coolant system pressures of either 2235 psig or 2085 psig. (The present full power allowed conditions are vessel average temperature of 567.8°F and pressure of 2235 psig.) No power uprating is being requested here. Thus, rated thermal power in T/S Definition 1.3 remains at 3250 Mwt. The program also supports 10% degradation of the safety injection and RHR pumps (but not the charging pumps) from the manufacturer's pump head curve, and 10% average steam generator tube plugging.

The transient and accident analyses performed by Westinghouse are, in general, intended to bound a potential future uprating in power for Unit 1. The non LOCA and large break LOCA accident analyses were performed at a core power level of 3413 Mwt, and covered a vessel average temperature range from 547°F to 576.3°F. The small break LOCA analysis, component evaluations and containment subcompartment studies were performed at a core power level of 3588 Mwt. These evaluations are intended to be bounding for both units. The assumed core power level of 3588 Mwt is consistent with the power level being considered for future power uprating for Unit 2 from its present rated thermal power of 3411 Mwt.

Although the majority of the analyses support a power increase and even an increase in Tavg, the area of operation supported by the RTP Program is limited to the present rated thermal power of 3250 Mwt, and upper limit on T_{avg} of 567.8°F. This is because the steam line break mass/energy release inside containment was not reanalyzed as part of the RTP Program. (The current analysis assumptions bound the RTP conditions.) Similarly, the steam line break core analysis assumed that the boron injection tank has been deleted and that the charging pump flow has been degraded 10%, but these cannot be incorporated because of the limitations imposed by the steam line break containment analysis.

The Westinghouse analyses and conclusions stand alone except for the following areas:

1. Containment Analysis

Sections 3.4.1.4, 3.7.2 and 3.7.3 of the Westinghouse report note that calculated peak pressures in various containment subcompartments due to postulated LOCAs may exceed the currently specified design pressures. AEPSC is evaluating the subcompartments to ensure that they can withstand the higher pressures predicted by the RTP Program. Preliminary analyses show that there is sufficient conservatism in the containment design to accomodate the increased pressure in most of the containment areas. Minor modifications may be required in some areas. These changes will be performed under our normal 10 CFR 50.59 design change process.

2. Steam Generator U-Bend Tube Fatigue Evaluation

Section 3.10.3.2 of the Westinghouse report noted that the fluid elastic instability, associated with the potential for steam generator U-tube vibration, will increase due to reduced temperature and pressure operation. U-tube vibration has become a concern in light of a recent tube rupture event at the North Anna Plant. Licensees were required by IE Bulletin 88-02 to evaluate the potential for vibration-induced fatigue cracking.

The Cook Nuclear Plant response to IE Bulletin 88-02 was transmitted via our letter AEP:NRC:1056 dated March 31, 1988. We concluded that vibration-induced fatigue was not a problem for the Cook Nuclear Plant because we have not detected any denting in unsupported tubes. Since the results of our evaluation were independent of the magnitude of the fluid elastic instability, the RTP Program does not impact the conclusion that vibration-induced fatigue is not a concern.

3. Reactor Coolant Piping and Supports

Section 3.10.6 of the Westinghouse report noted that an increase in reactor coolant system temperatures could create the possibility of interferences occurring between RCS components and equipment restraints due to thermal expansion. (The reactor coolant system is designed for free thermal expansion.) As discussed earlier, however, the RTP Program proposed in this letter does not include an increase in the present nominal value of Tavg. Thus, interference between the reactor coolant system and the restraints are not a concern at this time.

ATTACHMENT 3 TO AEP:NRC:1067

BALANCE OF PLANT EVALUATIONS AND

MISCELLANEOUS SAFETY EVALUATIONS

This attachment provides evaluations of the effect of the RTP program on balance of plant systems, as well as miscellaneous safety analyses that were outside the scope of the Westinghouse report.

BALANCE OF PLANT EVALUATIONS

The balance of plant evaluations considered the impact of operation of Unit 1 of the Cook Nuclear Plant at reduced temperatures and pressures only. The evaluation did not consider the impact of a potential future uprating in power and consequent increase in Tavg.

Spent Fuel Pool Cooling System (SFPCS)

The primary function of the SFPCS is to remove decay heat that is generated by the spent fuel pool elements stored in the pool. Decay heat generation is proportional to plant power level. Since the rated thermal power level of 3250 Mwt remains unchanged, the demands on the SFPCS are not increased. The purification function is controlled by SFPCS demineralization and filtration rates that are not affected by reduced primary pressures and temperatures.

Makeup Water System (MWS)

The MWS is designed for continuous service and is a shared system supplying demineralized water to both Unit 1 and Unit 2. The MWS contains three demineralizer trains each containing a cation exchanger, an anion exchanger, and a mixed bed demineralizer. There is a vacuum degasifier common to all demineralizer trains. Each train is designed for a maximum flow of 400 gpm and a normal flow of 300 gpm. The MWS demand is not affected by reduced primary pressure and temperatures. The MWS is adequately sized to handle small changes in normal flash tank blowdown rates currently limited by the orifice size.

Turbine Room Drainage System (Secondary Waste Liquid)

The purpose of the turbine room drainage system is to collect, neutralize and dispose of non-radioactive liquids from the turbine building through floor drains, equipment drains and roof drains. The turbine room drainage system consists of sumps and the associated piping and pumps. It has been determined that reduced primary temperature and pressure will not increase the volume of flow or flow rate of non-radioactive liquids in the turbine room drainage system; therefore, the operation of this system is not affected.

[illegible]

Nuclear Sampling System (NSS)

The NSS is designed to provide representative samples for laboratory analyses used to guide the operation of various primary and secondary systems throughout the plant during normal operation. Since reduction of sample pressure and temperature, when necessary, is already being done by heat exchangers and needle valves, it is determined that the parameters associated with the reduced temperature and pressure program do not affect the performance of the NSS.

Steam Generator Blowdown and Blowdown Treatment Systems

The steam generator blowdown system is used mainly to control secondary side water chemistry. It is also used to drain the steam generators during plant outages. The steam generator blowdown treatment system is used in the event of a primary to secondary steam generator tube leak to remove radioactive ions and particulates. This provides for continued blowdown usage while maintaining releases below 10 CFR 20 limits. Appropriate monitoring ensures that these limits are not exceeded. The maximum design flow rate of the blowdown treatment system is approximately 50 gpm.

Currently, the blowdown rate is controlled by the operator depending on system conditions. At a reduced primary temperature and pressure, the blowdown system will continue to perform its function as required to maintain the proper secondary side water chemistry. The treatment system will continue to operate based on the original design under a reduced primary temperature and pressure.

Chemical Feed Systems

The condensate and feedwater chemical feed systems supply the appropriate amount of chemical additive to the condensate and feedwater. The reduction of primary temperature and pressure will not significantly change the feedwater flow rate; therefore, the chemical feed systems will not be affected.

Auxiliary Feedwater System (AFS)

The AFS provides water to the steam generators when the main feedwater system is not available due to a loss of main feedwater, unit trip, feedwater or steam line break, loss of off-site power, or loss of coolant accident. The source of normal water is the condensate storage tank (CST) or the lake (emergency water source)

[illegible]

if the CST is unavailable. The AFS also provides water during start-up and shutdown when insufficient steam is available to drive the main feed pumps.

The AFS consists of one turbine-driven auxiliary feed pump, which feeds all four steam generators, and two motor-driven auxiliary feed pumps, each of which feeds two steam generators.

Westinghouse's RTP evaluation has shown that the AFS can supply the flow requirements for the various accidents, shown above, with the unit operating at RTP conditions. Therefore, RTP will not impact the existing AFS.

Containment Spray System (CTS)

The CTS provides spray cooling water to the containment atmosphere during a loss of coolant accident (LOCA) or a steam line break accident. This cooling water limits the peak pressure in the containment to below the containment design pressure (12 psig). A secondary function of the spray system is to remove radioactive iodine that would be released into containment during a break of the fuel cladding following a LOCA.

The effect of RTP on the containment pressure transient and its impact on containment integrity has been addressed by Westinghouse. The iodine removal capability of the containment spray system is not affected by the small pressure changes associated with RTP. No changes to the containment spray system are required due to RTP.

Component Cooling Water System (CCW)

The CCW system serves as an intermediate loop between reactor coolant or other potentially intermediate radioactive heat sources and lake water to ensure that leakage of radioactive fluid from components being cooled is contained within the plant. The CCW system is a closed cooling water loop consisting of a surge tank, a chemical addition tank and associated piping, valves, instrumentation, and the various equipment being cooled.

Since no power uprating is being requested at this time, the decay heat loads will not change. Therefore, the heat loads to the CCW system have not changed and RTP will not impact the existing CCW system.

Essential Service Water System (ESW)

The ESW system provides the cooling water requirements for the component cooling water heat exchangers, the emergency diesel generator coolers, the containment spay heat exchanges, and the control room air conditioning condensers. The ESW system, shared by both units, consists of four ESW pumps, each with an automatic backwashing duplex strainer and associated piping, valves, and instrumentation. The ESW system is comprised of two identical main headers. Each header is served by two pumps and each header, in turn, serves half of the system load in each unit.

The ESW system is operated in conjunction with the component cooling water and containment spray system that are not affected by the uprating. Therefore, the existing ESW system is not impacted with the unit operating at RTP.

Nonessential Service Water System (NESW)

The NESW system is a shared system that provides Lake Michigan water to be used as cooling and makeup water to numerous plant systems and components.

The system consists of four NESW supply pumps (two per unit) each with an automatic backwashing duplex strainer, cooling water suction supply lines from the circulating water intake and discharge tunnels, cooling water lines to the various components being serviced, and associated valves and instrumentation. The nonessential service water flows from the pumps to the equipment served and is then returned to the lake via the circulating water discharge tunnel.

Since no power uprating is required at this time, the heat loads from RTP conditions do not differ significantly from the present conditions. Therefore, the heat loads to the NESW system are not significantly affected and RTP will not impact the existing NESW system.

Feedwater Heater Extraction, Drains and Vents

The Unit 1 drains from the low pressure heaters No. 1, 2, 3 and 4 cascade to the external drains cooler, which drains to the main condenser. Drains from the No. 6 heater cascade to the shell of the No. 5 heater, which drains to the heater drain pumps. Two of three 50 percent capacity pumps inject this drain flow into the suction of the main feedpumps. Level control valves on the drain

lines of the low pressure heaters and on high pressure heater No. 6 automatically maintain the normal water level in the heaters. High pressure heater No. 5 drains by gravity only. Under RTP conditions the drain flows from heaters 1, 2, 3, and 4 will be slightly increased with no appreciable change in the amount of non-condensible gases. As a result of the small increase in drain flow, the level control valves will modulate further open to maintain the desired water level in the heaters. Setpoint changes for the level controllers will not be required. As the drain flow from heaters 5 and 6 will be reduced, the heater drain pump will maintain sufficient capacity. The vent system that removes the non-condensible gases from the shell side of the heaters will not be affected. Therefore, neither the drains nor venting system will require modification under RTP conditions.

Feedwater System (FS)

The FS consists of two feedpump suction strainers, two main feedpumps, two parallel strings of No. 5 and 6 high pressure heaters, four feedwater control valves and associated piping, valves and instrumentation.

The feedwater pumps take suction through the strainers from a common header supplied by the outlet flow from the low pressure heaters and the discharge from the heater drain pumps. Prior to being pumped to the steam generators, the feedwater is passed through the high pressure heaters where additional heat is added to the system.

RTP will not significantly impact the feedwater system. The slight changes in flow, pressure and temperature remain within the design limits of the system components.

Condensate System (CS)

The CS, in conjunction with the feedwater system, returns the condensed steam from the condensers and the feedwater heater drains to the steam generators while maintaining the overall water inventory throughout the cycle. The system is also required to compensate for the loss of fluid from the steam cycle when an atmospheric steam dump occurs. The necessary water inventory is maintained in the condenser hotwells and the condensate storage tank.

The condensate system for Unit 1 consists of one main condenser (separated into three shells, one for each low pressure turbine), three hotwell pumps, three condensate booster pumps, four steam

jet air ejectors, the turbine auxiliary cooling cycle, four stages of low pressure feedwater heating, the condensate storage tank and associated instrumentation, piping, and valves.

RTP will not impact the condensate system. With the steam flow into the condenser only slightly higher than pre-RTP conditions, the condensate temperature will remain basically the same. Therefore, system components will be unaffected.

Circulating Water System (CW)

The CW system is an open loop system that provides a heat sink for waste heat from the plant thermal cycle. The CW system supplies cooling water to various coolers and condensers during all phases of plant operation.

Lake Michigan water is piped into the forebay from which the CW pumps take suction. The pumps circulate the CW through the various services before it is returned to the lake.

RTP will not significantly impact the CW system. With a slightly higher steam flow into the main condenser the temperature rise across the condenser will be increased. However, the change in delta T is so slight that the CW system will not be affected.

Main Condenser Evacuation System

The steam jet air ejector system provides for air evacuation and removal of noncondensable gases from the main and feedpump turbine condensers to promote maximum efficiency from the condensers.

The steam jet air ejector system for each unit consists of three single element, single stage startup ejectors and four twin element, two-stage holding ejectors. The startup ejectors pull the vacuum in the condensers to approximately 15 inches Hg abs. From this point, the holding ejectors pull the condenser vacuum down to operating vacuum and maintain it at this level. After operating vacuum is obtained, the startup ejectors are no longer required and are shut down.

The steam jet air ejector system will not be impacted by RTP. With condenser conditions remaining essentially the same as pre RTP conditions, the current evacuation capacity, which includes 100 percent spare ejector capacity for the ejectors serving the main condensers, is more than adequate.

Turbine Auxiliary Cooling Water System (TACWS)

The TACWS is a subsystem of the condensate system. It uses main cycle condensate to remove heat from various heat exchangers associated with the turbine-generator unit.

The TACWS of each unit consists of two turbine auxiliary cooling pumps, one turbine auxiliary cooler and various other heat exchangers in the turbine-generator unit which are provided cooling water by the system.

RTP will not impact the TACWS. Condensate temperature and supply flow rates to the TACWS from the condensate system will not change. Heat loads from the turbine-generator unit heat exchangers will not increase under RTP conditions. Therefore, the existing TACWS will not require modification.

Main Turbine and Feed Pump Turbine

The main turbine consists of one double-flow HP turbine and three double-flow LP turbines. Reduced steam pressures at the inlet to the HP turbine will cause the unit to run with control valves in the wide open position and thus reduce throttling losses across the valves. There will be a slight increase in steam flow through the turbine leads. This 1.3% increase in flow will have negligible effect on velocities and pressure drops in these lines. This slight increase in flow through the HP turbine is due to the reduction in reheating steam to the moisture separator-reheater. Since this is a small change in flow, stage pressures in the turbine will be approximately the same with RTP operation. Increased moisture carryover will increase HP turbine maintenance. Steam generator modifications to maintain moisture at or below current design levels (0.25%) are being investigated to avoid this extra maintenance. Since the turbine will be running with valves wide open, a load reduction will be necessary to test valves. (There is no accurate method to predict the load at which to test the valves. The load must be established by trial and error method, e.g., reduce load to a conservative value of 90% and then test valves.)

Due to the reduced pressure in the steam generators, the feed pumps will require less power and the feed pump turbines (FPT) will require less steam flow to power the feed pump, which will run at a lower speed. Lower temperature reheat steam to the FPT will be offset by its higher density enabling the FPT to pass more mass resulting in little net change in FPT capacity.

Moisture Separator Reheater

With lower pressure in the main steam piping, reheating steam to the reheater is reduced. Reheating steam is taken off the main steam leads at the turbine bypass piping, provides single stage heating of the HP turbine exhaust steam, and drains to the #6 feedwater heaters through a drain tank level control. HP turbine exhaust flow to the moisture separator increases by less than 1%. Performance of the moisture separator-reheater will not be affected although the reheat temperature will be lower. Pressure drop increases will be insignificant. Increased main steam moisture carryover will tend to lower moisture separator effectiveness. This small reduction will have little effect on the rest of the cycle and steam generator modifications being investigated to reduce HP turbine maintenance due to increased carryover would also benefit moisture separator effectiveness.

This unit is designed to operate with the reheaters out of service. With reduced temperature and pressure operation, however, the unit will not be able to carry full load due to the flow passing capability of the control valves at the reduced pressure. The unit will be limited to about 95% rated output with RTP operation and reheaters out of service.

Turbine Steam Seal System (TSSS)

The TSSS provides sealing steam at locations on the turbine shaft where it passes through the casing. Its purpose is two-fold: 1) to prevent steam from leaking out along the shaft and into the atmosphere and 2) to prevent air from entering the turbine shells where vacuum in the cycle exists, as at the exhaust of the HP turbines. Pressures and enthalpies at each packing or gland change less than 1% due to RTP conditions. As such, seal flows will be approximately the same in both normal and RTP operation.

Main Steam System

This system consists of the piping from the steam generator to the turbine, turbine bypass piping, steam generator stop valves, safety valves, and power relief valves. The design flow in the steam leads is 14.236×10^6 lb/hr when operating at normal temperature and pressure in the primary system. With reduced temperature and pressure a slight reduction in steam flow to 14.198×10^6 lb./hr. is expected. With this operation, main steam pressure will reduce from 787 psia to 665 psia at the steam generator outlet. Changes in main steam moisture carryover mainly affect the HP turbine and moisture separator reheater and

[illegible]

are discussed in those sections. Velocities in the steam leads will increase by 16.6%, but remain within the recommended design velocity range. The steam generator safety valves were reviewed by Westinghouse and found capable of relieving the new flow rates.

The 6-inch power-operated relief valves are required to momentarily relieve pressure surges. They are capable of relieving 10% of the new flow rate, which is consistent with the present valve capacity.

There are four steam generator stop valves. Each steam generator lead contains one valve. The valves' function is to close within five seconds after receipt of an appropriate signal, thereby isolating the steam generators. The valves are designed to operate at 758 psi, 512^{o}F and a mass flow rate of 3,699,000 lbs/hr. The RTP operating conditions are 665 psi, 497.4^{o}F and a mass flow rate of 3,549,433 lbs/hr. Since the pressure, temperature, and mass flow rate have decreased, the steam generator stop valves will still be able to perform their intended function.

Steam Condition Effect on the Turbine Missile Analysis

The turbine missile analysis is based on an LP missile because the HP turbine casing will contain any potential missiles. Therefore, changes in HP turbine inlet steam conditions will have no effect on the missile analysis. The turbine manufacturer was contacted regrading the LP turbine whose inlet conditions do change. The conclusion was that the minor changes in steam conditions would not affect the missile analysis.

Ice Condenser Refrigeration System (ICRS)

The ICRS cools down the ice condenser from ambient conditions of the reactor containment and maintains the desired equilibrium temperature in the ice compartment. The ice condenser is sufficiently subcooled and insulated so that even a complete breakdown of the refrigeration or air handling system would not cause ice melting for one week.

The ICRS consists of 10 glycol chiller units, 6 glycol circulation pumps and 60 air handling units for each containment building. Since no increase in Tavg is requested, the containment heat load remains essentially unchanged. Therefore, the ICRS is unaffected by the RTP Program.

Waste Disposal System (WDS)

The WDS design is based on continuous operation of the primary plant with one percent defective fuel. Since no power uprating is requested at this time, the source term remains unchanged. It is therefore determined that the parameters associated with the reduced temperature and pressure program do not affect the performance of the WDS, nor its capability to collect potentially radioactive wastes discharged from the NSSS and process them as required to permit their removal or discharge from the plant site.

Refueling Water System (RWS)

The RWS provides a means of storage and transfer of refueling water required for refueling and fuel handling operations. It also functions as the source of water supply for the emergency core cooling and containment spray systems during post accident operations. The system consists of the refueling water storage tank (RWST), and associated piping, valves and instrumentation.

The Westinghouse RTP report concludes that the auxiliary tanks are not subject to any significant transients and therefore require no further evaluation. Additionally, the RTP T/S requirements for the RWST minimum contained volume is unchanged. Therefore, it is concluded that RTP will not impact the existing RWS.

Extraction Steam System (ESS)

The ESS is called the bleed steam system at the Cook Nuclear Plant. The bleed steam system for Unit 1 provides a source of steam to heat the condensate and feedwater. It also supplies steam to the auxiliary steam system.

At RTP conditions the bleed steam system will experience higher velocities due to a slightly higher mass flow rate. Calculations were performed which determined that velocities will not exceed standard industry guidelines for saturated steam flow. The increased velocities will cause only a minimal increase in erosion/corrosion.

Equipment and Floor Drainage System

The purpose of the equipment and flow drainage system is to collect and dispose of liquids from floors and equipment. The RTP Program will not significantly increase the flows, and therefore the system will not be significantly affected.

Fire Protection Systems and Fire Hazards Analysis

The fire protection systems and fire hazards analysis are independent of plant operating characteristics and are therefore not affected by the change.

HVAC Systems

The following HVAC systems were reviewed to determine the impact of reduced temperature and pressure operation. The HVAC systems will not be adversely affected, since the thermal heat loads either decrease or do not significantly increase as a result of the RTP program.

1. Auxiliary Building Ventilation
2. Containment Air Recirculation/Hydrogen Skimmer
3. Containment Purge
4. Fuel Handling Area Ventilation
5. Control Rod Drive Ventilation
6. Reactor Shield Ventilation
7. Diesel Generator Area Ventilation
8. Switchgear Room Ventilation
9. Service Water Intake Structure Ventilation
10. Containment Fan Cooling

Chemical and Volume Control System (CVCS)

The impact of RTP operation on the CVCS has been determined to be acceptable in the Westinghouse report. The program will not result in parameters outside the normal bounds of CVCS operation.

Turbine Bypass System

The turbine bypass system (steam dump) allows steam to bypass the turbine and go directly to the condenser. The steam dump system is physically sized to provide the capacity of an 85% loss of turbine or electrical generator capacity. The effects of RTP steam parameter changes would have minimal impact on this 85% capacity. However, maintenance concerns with the steam dump system lead us to consider reducing the capability of the system to 40%. This reduction to 40% capacity brings the Cook Nuclear Plant into conformance with most other Westinghouse PWRs. The accident analyses performed by Westinghouse do not take credit for the heat removal capability provided by the steam dump system.

[illegible]

Radiation Monitoring System

Since no power uprating is requested at this time, the source term is unchanged. Therefore, the radiation monitoring system is not affected by the RTP Program.

Radioactive Waste Treatment and Disposal Systems

Since no power uprating is requested at this time, the source term is unchanged. Thus, the RTP program does not adversely impact the radioactive waste treatment and disposal systems.

Secondary Systems Piping Supports

The change in pressures and temperatures in secondary systems is not significant and therefore has negligible impact on the piping support requirements for the secondary systems.

Electric Systems

Since the overall heat balance is not significantly changed, there is no impact on the BOP electrical systems. The only change to plant loads would be those associated with the reactor coolant pumps (RCPs). Required power for the RCPs is increased due to the increased water density at lowered Tavg conditions. Westinghouse has estimated that the pump temperature may rise no more than 7°C due to the increased horsepower. They have concluded that this temperature rise will not significantly affect motor life expectancy. We have reviewed cabling, penetrations, and the busses associated with the RCPs and have concluded they are adequate to handle the increased load.

MISCELLANEOUS SAFETY ANALYSES

The following accident analyses were outside the scope of the Westinghouse report, and were thus analyzed by AEPSC.

Hydrogen in the Containment following a Loss of Coolant Accident.

As a result of the rerating project, hydrogen accumulation in the containment following a LOCA was investigated. The hydrogen reanalysis was impacted by: (1) higher radiolysis rates due to assumed higher power levels, (2) increase of overall zirconium in the core to 50,000 lb from the previous value of 43,212 lb to allow for future core designs that may include greater amounts of zirconium, (3) increase in the amount of non-NSSS Aluminum allowed in containment, (4) removal of restrictions on location of

fast-corroding aluminum, (5) allowing the zirconium water reaction to produce significant amounts of hydrogen during the small break LOCAs as well as large break LOCAs, and (6) providing parametric sensitivity analyses for allowable subcompartment skimmer flows. The reanalysis was required at this time due to Items 1 and 5, and the other items were added to bound potential future operating conditions. The current analysis of record can be found in Chapter 14.3.6 of the Unit 2 updated FSAR.

The net result of these changes did not affect the results of the overall containment analyses (i.e., hydrogen concentrations were shown to be below 4% once the recombiners were turned on), but did affect the subcompartment analyses, in particular the pressurizer analyses.

Due to the fact that the pressurizer subcompartment is of small volume, a direct release of zirconium water reaction-produced hydrogen would result in volumetric concentrations greater than 4 v/o. In order to show that this was not a problem, a specific evaluation was made by Westinghouse of a postulated 3" break of the pressurizer spray line within that compartment. The selection of this break size followed evaluations which showed that this break would result in the highest rate of hydrogen generation of the 2", 3", and 4" breaks which were evaluated. For the 3" small break analysis, it was assumed that the zirconium water reaction began 20 minutes after the start of the accident and lasted for a total of 25 minutes.

Hydrogen generation rates accompanying the break were multiplied by a factor of over 30 as an added conservatism in order to account for uncertainties in break dynamics as well as selection of the design basis break.

Credit for steam dilution concurrent with the zirconium water reaction was taken. Prior to the onset of the zirconium-water reaction, specific quantitative analysis for steam mixing was not performed. Hydrogen concentrations during this time period would not be expected to exceed the 4 v/o limit, because of steam mixing/purging within the subcompartment.

After the zirconium water reaction had ceased, the adequacy of the specified purge rates were then determined by the concentrations of hydrogen calculated to occur. These calculations showed that if hydrogen skimmer flow was as low as 300 scfm, the calculated hydrogen concentration in the subcompartments would not exceed 4 v/o.

Similar studies were done for the skimmer flows associated with the fan/accumulator room, instrument room, and steam generator subcompartment, although due to the larger volumes, break specific dynamics were not involved. Each steam generator enclosure is interconnected with another, and they are separated by a wall with very large openings. Therefore, steam generator skimmer flow was not evaluated on an individual enclosure basis, but as an enclosure pair.

The results of these analyses showed that: (1) if skimmer flow were as low as 455 scfm from a pair of steam generators, hydrogen concentrations would not exceed 4 v/o, and (2) if skimmer flow from the instrument room were maintained above 70 scfm, and flows from the fan accumulator room were maintained above 91 scfm, hydrogen concentration below 4 v/o would be assured. These flows are within the capability of the containment recirculation/hydrogen skimmer system.

Input assumptions used for these reanalyses are presented in Tables 1 through 4. Results are presented in Tables 5 through 10. Rates of hydrogen generation are provided in Table 11. These hydrogen generation rates were provided by Westinghouse. Temperatures used in the analysis are presented in Table 12. The lower containment temperatures were provided by Westinghouse.

TABLE 1

PLANT PARAMETERS FOR CALCULATING
POST-ACCIDENT HYDROGEN GENERATION

Core Thermal Power	3600 MWt
Containment Free Volume	1.186×10^6 ft^3
Maximum normal operation containment temperature	120°F
Time at which recombiner starts	24 hours after LOCA
Recombiner flow rate	100 SCFM
Core and sump radiolysis assumptions	Regulatory Guide 1.7
Weight of zirconium cladding	50,000 lbs.
Percent of zirconium cladding reacted	1.5
Hydrogen produced by zirconium - water reaction	5925
Hydrogen from Reactor Coolant System	1,190 SCF
Aluminum inventory inside containment	Table 2
Zinc inventory inside containment	Table 3
Containment Spray pH	9.3
Aluminum corrosion rate	Figure 1
Zinc corrosion rate	Figure 2
Post-accident containment temperature transient	Table 12

TABLE 2

ALUMINUM INVENTORY INSIDE CONTAINMENT BUILDING

Component	Mass (pounds)	Surface Area (ft^2)
Nuclear Steam Supply System		
Source, Intermediate and Power Range Detectors	244	83
Flux Map Drive system	122	84
Miscellaneous Valve Parts	230	86
Reactor Coolant Pump Coolers	160	11,400
CRDM Connectors	193	42
Paint[2]	140	18,000
Air Handling Unit	44	18
Contingency[1]	250	85
Non-NSSS Items[3]		
Item that corrodes instantaneously	200	--
Item that corrodes throughout the transient	---	1,000

[1] contingency factor includes miscellaneous items such as valve handles, housing, etc.

[2] Paint on NSSS components is shielded from the sprays because these components are enclosed which are not directly sprayed.

[3] These items represent an upper limit estimate of total non-NSSS aluminum.

TABLE 3

ZINC INVENTORY INSIDE CONTAINMENT BUILDING

Ice Condenser Related	Weight (lbs)	Surface Area (ft^2)
Ice Baskets	113,415	453,660
Ice Condenser Walls	738	29,500
Non Ice Condenser Related		
Upper Limit on Zinc Surface Area	------	350,000*

*The analysis has conservatively assumed 350,000 square feet of zinc in the containment (non ice condenser related). No limit was set as to the thickness of the zinc and the analysis conservatively allowed the 350,000 ft^2 of zinc non ice condenser related to corrode throughout the transient.

TABLE 4

FRACTION OF EACH HYDROGEN CONTRIBUTION CONSIDERED FOR SUBCOMPARTMENT ANALYSIS

Subcompartment	Sump Radiolysis	Core Radiolysis	Zirconium Water Reaction (2)	Corrosion
Steam Generator (1)	.10	1.00 (2)	1.00	See Note (7)
Pressurizer	.05	1.00 (2)	See Note (3)	See Note (4)
Fan Accumulator Room	.15	See Note (5)	1.00	See Note (6)
Instrument Room	.12	See Note (5)	1.00	See Note (6)

Notes:

(1) Hydrogen concentration based on skimmer flows, hydrogen generation rates, and volume of a steam generator enclosure pair.

(2) Hydrogen from zirconium-water reaction assumed to be distributed thoroughout entire lower volume, as is hydrogen release from first two minutes of core radiolysis.

(3) Conservatively based on assumed 3-inch break in pressurizer subcompartment.

(4) Hydrogen generated due to corrosion of Aluminum and Zinc based on actual inventory of these corrodable metals in the pressurizer compartment.

(5) 100% of core radiolysis distributed uniformly throughout lower volume.

(6) Hydrogen generated as a result of metal corrosion (zinc and aluminum) is assumed to be distributed in proportion to the volume of the subcompartment.

(7) Hydrogen generated due to corrosion of Aluminum and Zinc based on actual inventory of these corrodable metals in the steam generator doghouse.

TABLE 5

HYDROGEN CONCENTRATION IN
STEAM GENERATOR SUBCOMPARTMENT PAIR

Time (min)	500 scfm		450 scfm		400 scfm	
	scf	v/o	scf	v/o	scf	v/o
10	655.1	3.03	655.1	3.03	655.1	3.03
20	803.5	3.72	798.1	3.69	796.6	3.68
30	829.1	3.83	829.9	3.84	839.7	3.87
40	836.3	3.87	849.3	3.93	866.8	4.01
50	840.4	3.89	860.0	3.98	884.3	4.09
60	840.0	3.88	870.1	4.02	895.1	4.14
70	837.2	3.87	870.9	4.03	901.6	4.17
80	833.1	3.85	869.6	4.02	904.8	4.18
90	828.3	3.83	866.8	4.01	905.8	4.19
100	825.7	3.82	861.4	3.98	904.1	4.18

TABLE 6

HYDROGEN CONCENTRATION IN
PRESSURIZER SUBCOMPARTMENT

Time (min)	500 scfm		400 scfm		300 scfm	
	scf	v/o	scf	v/o	scf	v/o
10	132.5	(1)	132.5	(1)	132.5	(1)
20	111.1	(1)	125.5	(1)	143.7	(1)
	Start Zr-H_2O Reaction					
25	22.4	(2)	(3)	(2)	(3)	(2)
30	22.2	(2)	(3)	(2)	(3)	(2)
35	46.4	(2)	(3)	(2)	(3)	(2)
40	21.0	(2)	(3)	(2)	(3)	(2)
45	8.49	(2)	8.49	(2)	8.49	(2)
	Stop Zr-H_2O Reaction					
50	57.2	2.05	52.0	1.87	50.1	1.80
60	89.2	3.20	86.9	3.12	92.0	3.30
70	88.8	3.19	94.1	3.38	105.3	3.78
80	88.1	3.16	95.3	3.42	109.5	3.93
90	88.1	3.16	95.5	3.43	110.7	3.98
100	87.3	3.14	94.5	3.39	109.7	3.94
110	87.0	3.12	93.8	3.37	108.7	3.90
120	87.8	3.15	94.4	3.39	108.9	3.91
130	--	--	--	--	107.8	3.87
140	--	--	--	--	106.7	3.83

Notes

(1) Compartment assumed full of steam, but no credit for steam dilution in calculating scf.

(2) Compartment full of steam, credit taken for steam dilution in calculating scf.

TABLE 7

HYDROGEN CONCENTRATION IN
FAN ACCUMULATOR ROOM

Time	100 scfm	95 scfm	90 scfm
10 min.	662 scf	662 scf	662 scf
1 hr.	762 scf	765 scf	769 scf
5 hr.	857 scf	873 scf	891 scf
6 hr.	858 scf (3.83 v/o)	876 scf	895 scf
7 hr.	858 scf	877 scf (3.91 v/o)	897 scf
8 hr.	857 scf	877 scf	898 scf
9 hr	858 scf	877 scf	899 scf (4.01 v/o)
10 hr.	840 scf	859 scf	880 scf
11 hr.	--	--	862 scf

TABLE 8

HYDROGEN CONCENTRATION IN
INSTRUMENT ROOM

Time	100 scfm	80 scfm	60 scfm
10 min.	413 scf	413 scf	413 scf
5 hr.	465 scf (3.36 v/o)	515 scf (3.72 v/o)	478 scf
6 hr.	461 scf	511 scf	579 scf
7 hr.	458 scf	508 scf	579 scf (4.12 v/o)
8 hr.	457 scf	506 scf	578 scf
9 hr.	458 scf	--	576 scf
10 hr	449 scf	--	563 scf
11 hr.	--	--	550 scf

TABLE 9

HYDROGEN CONCENTRATION,
LOWER AND UPPER VOLUME

Time (min)	Lower Active Volume scf	Lower Active Volume v/o	Upper Volume scf	Upper Volume v/o
10	9623	3.79	1138	0.11
15	4976	2.09	6047	0.53
20	3185	1.38	8043	0.79
25	2537	1.18	8856	0.87
30	2369	1.10	9184	0.90
45	2336	1.12	9686	0.95
60	2381	1.15	10,084	0.99
75	2456	1.19	10,432	1.02

TABLE 10

OVERALL CONTAINMENT HYDROGEN CONCENTRATIONS; 1.5% $Zr-H_2O$ REACTION; RECOMBINER STARTS AT 24 HRS. AFTER ACCIDENT

Time	scf of H_2	v/o of H_2
2 min.	9271	0.93
1 hr.	11,382	1.14
6 hr.	18,058	1.81
17 hr.	25,486	2.55
24 hr.	27,718	2.78
Start Electric Hydrogen Recombiner		
27 hr.	28,813	2.88
2 day	30,361	3.03
3 day	31,536	3.15
4 day	32,208	3.21
5 day	32,699	3.25
6 day	33,003	3.28
6 day, 6 hr.	33,021	3.28
6 day, 12 hr.	33,003	3.28

TABLE 11

RATES OF HYDROGEN GENERATION

TIME	SUMP RADIOLYSIS (scfm)	CORE RADIOLYSIS (scfm)	CORROSION (scfm)
10 min	15.2	8.61	29.3
20 min	14.2	7.09	11.7
30 min	13.4	6.20	11.7
40 min	12.7	5.61	11.7
50 min	12.1	5.18	11.7
60 min	11.6	4.85	11.7
1 hour	11.6	4.85	11.7
6 hour	5.33	2.85	9.0
12 hour	3.10	2.22	4.34
18 hour	2.11	1.90	1.39
24 hour	1.62	1.72	1.39
1 day	1.62	1.72	1.39
2 day	0.935	1.43	1.03
3 day	0.695	1.29	1.03
4 day	0.573	1.21	1.03
5 day	0.504	1.15	1.03
6 day	0.459	1.10	1.03
7 day	0.426	1.07	1.03

TABLE 12

TEMPERATURES USED IN HYDROGEN ANALYSIS

TIME SPAN	LOWER CONTAINMENT TEMPERATURE (°F)	UPPER CONTAINMENT TEMPERATURE (°F)
0 to 40 seconds	240	120
40 to 100 seconds	230	120
100 to 300 seconds	220	120
300 to 1,000 seconds	204	120
1,000 to 7,000 seconds	180	120
7,000 to 10^5 seconds	173	173
greater than 10^5 seconds	153.1	153.1

DONALD C.COOK NUCLEAR PLANT

CORROSION RATE OF ALUMINUM AS A FUNCTION OF TEMPERATURE

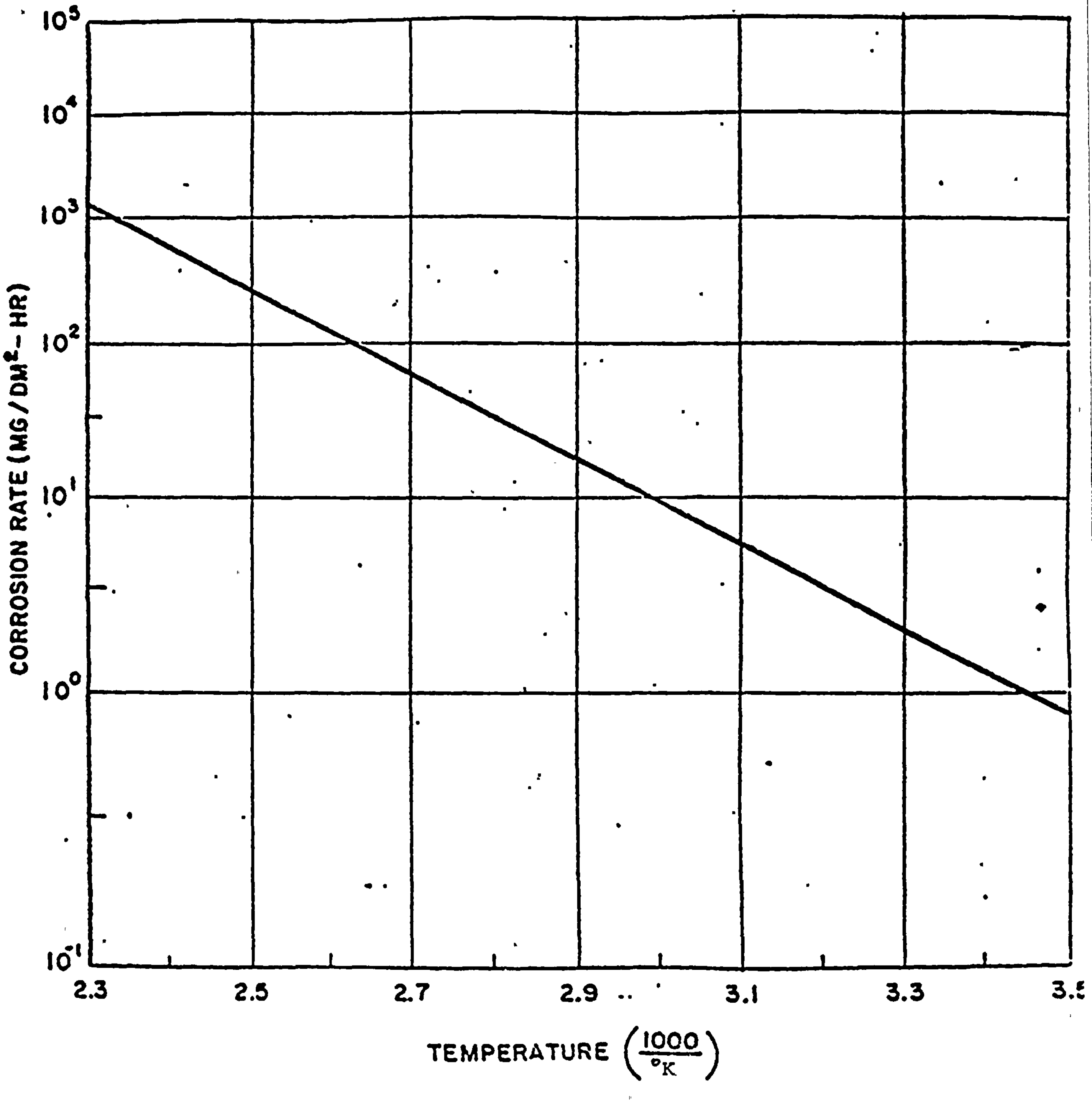

FIGURE 1

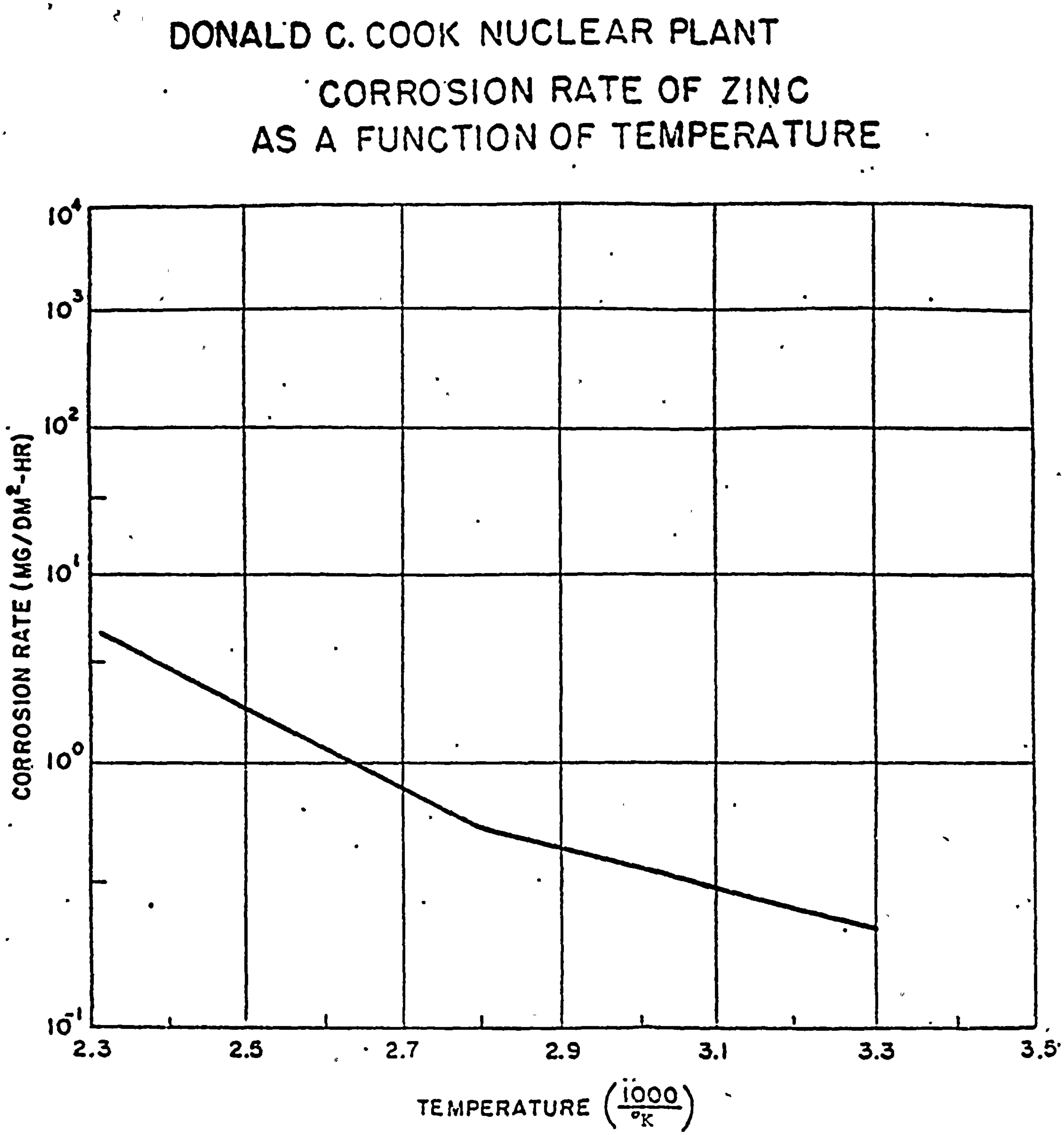
DONALD C. COOK NUCLEAR PLANT
CORROSION RATE OF ZINC
AS A FUNCTION OF TEMPERATURE
CORROSION RATE (MG/DM²-HR)
10⁴
10³
10²
10¹
10⁰
10⁻¹
2.3
2.5
2.7
2.9
3.1
3.3
3.5
TEMPERATURE (1000/°K)

FIGURE 2

[illegible]

Environmental Qualification Analyses

A. High Energy Line Break Outside Containment

These analyses are primarily associated with high energy line breaks (HELBs) outside of containment. High energy systems are defined as those having a normal service temperature above 200°F, a normal operating temperature above 275 psig and a nominal diameter greater than one inch. The systems at the Cook Nuclear Plant that fall under the above definition are the main steam system, the feedwater system, the steam generator blowdown system, the chemical and volume control system, and the steam supply to the auxiliary feedwater pump turbine.

Design basis breaks are defined in the FSAR. The consequences of each break were evaluated as described in the FSAR for each system with respect to (1) high compartment differential pressure, (2) jet effects, (3) potential pipe whip damage and (4) environmental effects. All assessments of potential damage were expressed in terms of the postulated breaks not causing any damage to any of the equipment required to assure safe shutdown of the plant. Jet impingement and pipe whip forces were calculated based on the maximum operating pressure. The compartment differential pressures and environmental effects (pressure and temperature profiles) are dependent on the break mass and energy release rates. The design/initial conditions for each system have been reviewed against the new operating conditions to determine if the new operating conditions are bounded by the existing analysis.

In both the Westinghouse analyses given in Attachment 2 and the balance of plant evaluations described earlier, main steam pressures are either the same or lower for the reduced temperature and pressure conditions. Therefore, the pressure conditions for the steam generator blowdown system or the steam supply to the auxiliary feedwater pump turbine would also be the same or lower. As stated in the balance of plant evaluation, changes in the feedwater system pressure are within the design limits of the system. The chemical and volume control system (CVCS) was discussed in the Westinghouse analyses. The CVCS will operate at the same or a lower pressure as before. Therefore, based on this review, the conditions assumed in the HELB analyses are still bounding for the reduced temperature and pressure conditions, and the mass and energy release rates are still applicable.

B. Flooding

Flooding in the auxiliary building due to a failure of non-seismic Class I piping has been reviewed. Only systems having access to large water volumes and/or potentially large flowrates were considered as discussed in the FSAR. The only such system is the main feedwater system. Since the changes in flow in the feedwater system are still within the design limits, the results discussed in the FSAR are still applicable.

Flooding in containment is slightly increased due to the larger initial water mass in the reactor coolant system because of the higher density at the reduced temperature. This change was found to be within the volume margins used to determine the maximum flood-up elevation.

C. Environmental Qualification Inside Containment

The various analyses contained in the Westinghouse report and in the containment long term pressure analysis submitted previously (WCAP 11908) were reviewed with regards to the Environmental Qualification parameter envelope. In all cases the parameters resulting from the RTP program were within the bounds of our present envelope, and therefore no changes are necessary.