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INDIANA
MICHIGAN
POWER

September 30, 2000

C0900-04
10 CFR 50.90

Docket No.: 50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
LICENSE AMENDMENT REQUEST
STEAM GENERATOR TUBE SURVEILLANCE INTERVAL EXTENSION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Unit 2 Facility Operating License, DPR-74. I&M proposes to add a condition to DPR-74 to allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification (T/S) surveillance requirement 4.4.5.3. The surveillance is required to be completed by March 5, 2001, which would follow less than 10 months of operation since the last inspection. I&M proposes to allow the inspection to be performed during the refueling outage following the current operating cycle, cycle 12, but no later than June 30, 2002. Operating cycle 12 is projected to end in March of 2002. This change will avoid a mid-cycle shutdown, which would be required to meet the 40-calendar month inspection interval of T/S 4.4.5.3. The CNP Unit 2 steam generators were replaced in 1988 with steam generators having improved tube material. Given the operating history and material condition of these steam generators, this extension does not involve any significant increase in risk.

Attachment 1 provides a detailed description and safety analysis to support the proposed change. Attachment 2 provides marked-up Facility Operating License pages for Unit 2. Attachment 3 provides the Unit 2 Facility Operating License pages with the proposed changes incorporated. Attachment 4 describes the evaluation performed in accordance with 10 CFR 50.92(c), which concludes that no significant hazard consideration is involved. Attachment 5 provides the environmental assessment. Attachment 6 contains an assessment of the impact of the Unit 2 extended shutdown period on the continued operability of the steam generators. Attachment 7 lists the commitments made in this submittal.

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To allow for changes in outage planning, I&M requests approval of this request by November 15, 2000. A 30-day implementation period is proposed.

No previous submittals affect this request. One commitment was identified in this submittal:

- I&M will have the Electric Power Research Institute perform an independent review of the 1997 Cook Nuclear Plant Unit 2 steam generator tube inspection techniques and results. I&M will report any adverse findings from this review to the Nuclear Regulatory Commission.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs, at (616) 697-5056.

Sincerely,

for R. P. Powers
Vice President

/dmb

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, Michael W. Rencheck, being duly sworn, state that I am Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

MW Rencheck

M. W. Rencheck
Vice President Engineering

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 30 DAY OF September, 2000

Danielle M. Schrader
Notary Public

My Commission Expires 4-4-2004

DANIELLE M. SCHRADER
Notary Public, Berrien County, MI
My Commission Expires Apr 4, 2004

ATTACHMENT 1 TO C0900-04

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGE

A. Summary of the Proposed Change

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Unit 2 Facility Operating License, DPR-74. I&M proposes to add a condition to DPR-74 to allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification (T/S) surveillance requirement 4.4.5.3. The surveillance is required to be completed by March 5, 2001, which would follow less than 10 months of operation since the last inspection. I&M proposes to allow the inspection to be performed during the refueling outage following the current operating cycle 12, or no later than June 30, 2002. Operating cycle 12 is projected to end in March of 2002. This change will avoid a mid-cycle shutdown, which would be required to meet the 40-calendar month inspection interval. The CNP Unit 2 steam generators were replaced in 1988 with steam generators having improved tube material. Given the operating history and material condition of these steam generators, this extension does not involve any significant increase in risk.

B. Description of the Current Requirement

T/S surveillance requirement 4.4.5.3 specifies the required inspection intervals for in-service inspection of steam generator tubes.

The Unit 2 steam generators meet the prerequisite conditions to allow the inspection interval to be extended to once per 40 months. Specifically, the last two tube inspections have followed all-volatile treatment conditions and all results were in category C-1, meaning less than 5% of the tubes inspected were degraded and none of the inspected tubes were defective. Based on the November 5, 1997, completion date of the last inspection, the next inspection is required to be completed by March 5, 2001.

C. Bases for the Current Requirement

The basis for the steam generator tube surveillance requirements is to provide assurance that the structural integrity of this portion of the reactor coolant system (RCS) is maintained. The specified interval ensures tube degradation is detected and monitored in a manner that assures safe and reliable operation between inspections. The basis for the primary-to-secondary leak rate T/S limit of 500 gallons per day in any one steam generator is to provide assurance of steam generator tube integrity in the event of a main steam line rupture or loss-of-coolant accident. The total T/S leakage limit of one gallon per minute from all four steam generators ensures that the dose contribution from tube leakage will be limited to a small fraction of 10 CFR 100 limits in the event of either a steam generator tube rupture or main steam line break.

In-service inspection of the steam generator tubing is necessary to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

D. Need for Revision of the Requirement

Due to the extended Unit 2 shutdown period from September 1997 to June 2000, the projected end-of-cycle does not coincide with the required inspection interval. The surveillance due date of March 5, 2001, will require a mid-cycle shutdown to meet the 40-calendar month interval.

E. Description of the Proposed Change

I&M proposes to add a condition to Operating License DPR-74 to allow an extension of the Unit 2 steam generator tube inspection surveillance requirements of T/S 4.4.5.3. The proposed amendment would allow the next inspection to be completed during the next refueling outage, but no later than June 30, 2002. The next refueling outage is expected to begin in March of 2002. The proposed change would extend the 40-month surveillance interval to a maximum one-time interval of 56 calendar months. This is the first surveillance interval extension request for these steam generators since they were installed.

F. Bases for the Proposed Change

Electric Power Research Institute (EPRI) pressurized water reactor steam generator examination guidelines recommend an interval based on effective full power months or refueling outage cycles, rather than an absolute time interval. Based on the T/S required interval, the next inspection is required to be performed by March 5, 2001, when Unit 2 will have accumulated less than 10 months of power operation since the last inspection. Under the proposed extension, EPRI guidelines will continue to be met by performing inspections during the refueling outages before and after Cycle 12. Additional details regarding recent inspection results and program conformance are included in Attachment 6. The proposed inspection deadline of June 30, 2002, is based on an estimate of the latest date of the next refueling outage.

The Unit 2 steam generators were replaced in 1988. A pre-service inspection including all tubes was performed and additional tube inspections were performed in 1990, 1992, 1994, and 1997. While a total of 15 tubes have been plugged in the Unit 2 steam generators since their installation in 1988, none were due to corrosion-related degradation. One tube was plugged in 1988 due to a manufacturing defect discovered during the pre-service inspection. Nine tubes were plugged in 1994 because of mechanical damage from off-line maintenance activities. Five tubes were plugged in 1997 due to foreign object-induced wear indications. These 15 tubes correspond to 0.10% of the total number of tubes, which is well below the licensed plugging limit of 10%. Based on the results

of the last inspection, discussed in Attachment 6, and the lack of operational time, it is expected that there will be no steam generator tubing degradation concerns before the next refueling outage.

The Unit 2 steam generators incorporate enhanced material and design features. The tubes are constructed of corrosion-resistant alloy 690. Industry experience has demonstrated that alloy 690 has significantly enhanced corrosion resistance over its alloy 600 predecessor. Other design enhancements include an increased row 1 U-bend radius and heat-treated low row U-bends. The tube support structure consists of one octofoil flow distribution plate, seven support plates with quatrefoil holes, and six sets of anti-vibration bars. The flow distribution plate, support plates, and anti-vibration bars are made from corrosion-resistant type 405 stainless steel. These features considerably reduce the susceptibility to degradation. I&M considered recent industry experience regarding concerns with steam generator tube inspections, and concluded that these do not apply to the CNP Unit 2 steam generators. Industry issues are addressed in Attachment 6. An independent review was performed of the Unit 2 Chemistry program for the four-year period including the previous operating cycle and subsequent shutdown. Details of the chemistry review are provided in Attachment 6.

No active degradation mechanisms, based on EPRI definitions, are present in any of the four steam generators. Based on past inspection results, the steam generators have experienced negligible degradation, as evidenced by an overall plugging percentage of 0.10%. The most limiting imperfections that were detected in tubing that remains in service are wear indications of no more than 11% through-wall. These indications would not change while the steam generators are out of service. Assuming conservative growth rates, these indications will not approach plugging limits for several additional cycles of operation. As of September 26, 2000, the calculated steam generator leak rate was zero gallons per day after 99 days of Mode 1 operation.

Sludge lancing was performed in 1997. Subsequent visual inspections determined the tube sheet area to be free of sludge and verified the overall good condition of select secondary side areas. Throughout the shutdown period, the steam generators did not experience the temperature and pressure conditions typically associated with corrosion-based degradation. No chemical excursions that would result in rapid steam generator corrosion occurred during the previous operating cycle, as both primary and secondary chemistry were well maintained. This provides further confidence that degradation has been minimal and is not expected to be a concern for the current operating cycle.

The extended unit shutdown is not expected to have degraded the stainless steel support plates or other steam generator internals. The lack of power operation minimizes service-based degradation such as wear. No degradation concerns are expected due to the layup environment that has been maintained in the steam generators.

Several other plants, including CNP Unit 1, have been granted similar requests in recent years. Applications approved in 1999 include Beaver Valley 1, Farley 1, and Crystal River 3. These

plants lack the material enhancements, such as alloy 690 tubing, that are present in the CNP Unit 2 steam generators, and have experienced significantly more tube degradation. An interval extension for the CNP Unit 1 tube surveillance was granted in December 1998.

G. Discussion of Risk

While the time between tube inspections would be more than the currently allowed T/S interval of 40 months under the proposed extension, the actual operating time between inspections would be only approximately 18 effective full power months. This is half of the operating time between inspections that would normally be allowed by the 40-month interval. The extended period spent under layup conditions with favorable temperature, pressure, and chemistry conditions is considerably less conducive to tube degradation than an additional operating cycle.

The accident considered directly applicable to this proposed surveillance interval extension is a steam generator tube rupture. This accident is addressed in Updated Final Safety Analysis Report section 14.2.4. Based on the materials and condition of the Unit 2 steam generators, as discussed in Attachment 2, the proposed one-time extension of the surveillance interval is not expected to have an effect on the steam generator tube rupture or any other accident analysis.

The tube plugging limit of 40% nominal tube wall thickness represents a 60% margin before tube leakage. As described in Nuclear Regulatory Commission Information Notice 91-43, "Recent Incidents Involving Rapid Increases in Primary-To-Secondary Leak Rate," tubes typically leak before failing, so leak rate monitoring can reduce the probability of a steam generator tube rupture.

There are no tubes in service in any of the Unit 2 steam generators that were found to be degraded (20% or more through-wall) in any previous inspection. Surveillance requirement 4.4.5.4 requires a minimum of 3% of the total number of tubes to be inspected if the steam generators are in degradation category C-1. The Unit 2 steam generators have been in degradation category C-1 in every inspection since they were installed in 1988. During the last inspection, 50% of the tubes were inspected, and none were found to exceed the plugging limit.

The lack of tube degradation, along with the material and design features and chemistry controls, as addressed in Attachment 6, provides reasonable assurance that tube repair limits will not be approached during the current operating cycle. Therefore, the probability of steam generator tube rupture is not significantly increased by the proposed change. Additionally, RCS activity is not affected and leakage rate limits are not changed by the proposed extension, so the consequences of an accident are not increased. Because there is no significant increase in the probability or consequences of an accident, the proposed extension does not involve any significant increase in risk.

ATTACHMENT 2 TO C0900-04

FACILITY OPERATING LICENSE PAGE
MARKED TO SHOW PROPOSED CHANGE

2.C.(7) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall be described in the station chemistry manual and shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

Amendment No. 18

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

2.C (8) The steam generator tube inspection surveillance requirements of Technical Specification 4.4.5.3 have been extended until the start of cycle 13, but no later than June 30, 2002.

D. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Donald C. Cook Nuclear Plant Security Plan," with revisions submitted through July 21, 1988; "Donald C. Cook Nuclear Plant Training and Qualification Plan," with revisions submitted through December 19, 1986; and "Donald C. Cook Nuclear Plant Safeguards Contingency Plan," with revisions submitted through June 10, 1988. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

Amendment No. 109

E. Deleted by Amendment 63.

F. Deleted by Amendment 6.

ATTACHMENT 3 TO C0900-04

PROPOSED REVISED FACILITY OPERATING LICENSE PAGE

2.C.(7) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall be described in the station chemistry manual and shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

Amendment No. 18

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

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Amendment No. 109

E. Deleted by Amendment 63.

F. Deleted by Amendment 6.

ATTACHMENT 4 TO C0900-04

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has evaluated this proposed amendment and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license does not involve a significant hazard if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

I&M proposes to amend the Unit 2 Facility Operating License, DPR-74. I&M proposes to add a condition to DPR-74 to allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification (T/S) surveillance requirement 4.4.5.3. The surveillance is required to be completed by March 5, 2001, which would follow less than 10 months of operation since the last inspection. I&M proposes to allow the inspection to be performed during the refueling outage following the current operating cycle, cycle 12, but no later than June 30, 2002. Operating cycle 12 is projected to end in March of 2002. This change will avoid a mid-cycle shutdown, which would be required to meet the 40-calendar month inspection interval. The CNP Unit 2 steam generators were replaced in 1988 with steam generators having improved tube material. Extending the steam generator inspection surveillance requirement to the end of operating cycle 12 will allow only approximately 18 full power months between inspections. Given the operating history and material condition of these steam generators, this extension does not involve any significant increase in risk.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The accident considered applicable to the proposed change is a steam generator tube rupture (SGTR). The precursors/initiators of a SGTR (degraded, defective, or leaking tubes) are not known or expected to be present in the CNP Unit 2 steam generators. These steam generators were newly installed in 1988, and include corrosion prevention design features not included in previous generations of steam generators.

There are no active degradation mechanisms present in the Unit 2 steam generators. Any tube imperfections that may be present or that may be initiated during the current operating cycle are not expected to progress to the point of tube failure before the next refueling outage.

Considering the condition of the steam generators and the operational time between inspections, the proposed change will not significantly increase the probability of occurrence of an accident.

The proposed change will not affect the scope, methodology, acceptance limit, or corrective measures of the existing steam generator examination program.

Unit 2 recently completed an extended shutdown that effectively limited the operational time that is the basis for the surveillance frequency. When the reactor is shut down and the reactor coolant system is at a reduced temperature, the steam generators are not subject to conditions that lead to significant tube degradation. Based on power operation time, the proposed extension will not increase the operating interval between surveillances beyond that currently allowed by T/S.

The steam generator tube inspection interval is not used in the SGTR accident analysis. The proposed change will, therefore, not affect the accident analysis or methodology.

The severity of an analyzed tube rupture event is not related to the time interval between inspections. The proposed change does not affect allowable leakage rates or source terms, and does not change the duration of an SGTR or the response to the event. Because the severity of an accident is not increased by the proposed change, there is no impact on offsite dose considerations.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will not result in a change in plant configuration or operation. Plant systems and components will not be operated in a different manner because of this change. The proposed change does not affect or create new accident initiators or precursors.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The T/S limit of one gallon per minute total steam generator tube leakage ensures the offsite dose from tube leaks is limited to a small fraction of 10 CFR 100 limits. The T/S leakage limit of 500 gallons per day in one steam generator is based on ensuring tube integrity in the event of a steam line rupture or loss of coolant accident. Because the offsite dose considerations from steam generator tube failures are limited by the primary-to-secondary leak rate program and not the tube inspection program, the proposed change has no impact on offsite dose.

There are no tubes in service in any of the Unit 2 steam generators that were found to be degraded, and no active steam generator tube degradation is known to be occurring. Therefore, the available margin in tube wall thickness is not being significantly reduced. During the last inspection, 50% of the tubes were inspected (more than sixteen times the T/S requirement), and none were found to exceed the plugging limit, providing additional assurance that safety margins are not being reduced. The absence of tube degradation, along with the material and design features and chemistry controls, provide reasonable assurance that tube repair limits will not be approached during the current operating cycle.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no significant hazards consideration.

ATTACHMENT 5 TO C0900-04

ENVIRONMENTAL ASSESSMENT

Indiana Michigan Power Company (I&M) has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 4, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed amendment involves only an extension to the steam generator tube inspection interval, and does not result in the generation of any additional radioactive or nonradioactive effluents. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in significant changes to the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

ATTACHMENT 6 TO C0900-04

UNIT 2 STEAM GENERATOR CYCLE 12 REVIEW

1.0 Introduction

Since the completion of the 1997 steam generator inspection, the resolution of unrelated design basis issues delayed Unit 2 restart for an extended period. The startup delay resulted in reducing the allowable runtime during the calendar-based inspection interval such that a full fuel cycle cannot be completed before the next technical specification (T/S) required inspection. Under the current requirements, Unit 2 will accumulate less than 10 months of operating time before requiring a forced shutdown to comply with the 40-month inspection interval. The proposed amendment would allow the Unit 2 steam generators to complete one operating cycle of approximately 18 full-power months between inspections.

For steam generators with no active degradation mechanisms, which include the Donald C. Cook Nuclear Plant (CNP) Unit 2 steam generators, the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) Steam Generator Examination Guidelines recommend performing tube inspections either at each refueling outage, or a larger scope inspection every other refueling outage. The EPRI guideline is incorporated by reference in Nuclear Energy Institute (NEI) document NEI 97-06, "Steam Generator Program Guidelines." The proposed interval extension will not conflict with either of these guideline documents.

2.0 T/S Requirements

T/S surveillance requirement 4.4.5.3 defines the frequency of steam generator eddy current tube inspections. This T/S allows the inspection interval to be extended up to 40 calendar months provided the steam generators have been under all-volatile technology conditions and all inspection results from the last two inspections were in the C-1 category, or if two consecutive inspections indicate that previously observed degradation has not continued and no additional degradation has occurred.

The C-1 category corresponds to inspection results with less than 5% of the inspected tubes degraded and none of the inspected tubes defective. The T/S define a degraded tube as a tube containing imperfections at or above 20% of the nominal wall thickness caused by degradation. Degradation is defined as a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube. Defect is defined as an imperfection (exception to dimensions, finish or contour of a tube from fabrication drawings or specifications) of such severity that it exceeds the plugging limit (40% reduction in nominal tube wall thickness).

The last two Unit 2 eddy current inspections, conducted in 1994 and 1997, resulted in two consecutive inspections in the C-1 category.

In 1994, no in-service degradation was noted during the 6% random eddy current inspection that was conducted on two of the four steam generators. However, nine tubes were plugged due to mechanical damage that occurred during off-line secondary side pressure pulse cleaning. The nine tubes were preventively plugged and were not considered degraded since the damage was not service-induced. The tubes were also not defective, since they did not violate the 40% plugging limit. The steam generators were therefore in degradation category C-1.

In 1997, one degraded tube was identified. The degraded tube contained a 28% reduced wall thickness indication at a cold leg support. Eddy current analysis determined the degradation to be a result of wear caused by the presence of a burr or small foreign object lodged near the center of the quatrefoil support. This tube was preventively plugged due to a lack of growth rate information. This tube was not considered defective since the degradation did not exceed the 40% reduced wall thickness plugging limit. As over 7,100 tubes were examined in the base inspection, the single degraded tube represents much less than the 5% allowed for a C-1 classification.

During the 1997 inspection, four additional tubes were plugged in one steam generator because of outside diameter volumetric indications at the top of the tubesheet. Eddy current analysis could not assign a percentage through-wall value to these indications due to the lack of a qualified sizing technique. As a conservative measure, the tubes were plugged. The tubes were not considered degraded or defective since the damage was representative of foreign object-induced wear and was not a result of progressive degradation due to design, manufacturing errors or typical in-service conditions. Expanded eddy current inspections and foreign object search and retrieval operations were conducted, resulting in the removal of a foreign object (weld wire) from the affected steam generator. As no other indications of degradation or defective tubes were identified, the 1997 inspection results were categorized as degradation category C-1.

3.0 Steam Generator Description

Design Features

Unit 2 contains four Westinghouse model 51F replacement steam generators. The original steam generators were replaced in 1988 due to accelerated tube degradation. The replacement steam generators were designed and manufactured with specific material and design enhancements to reduce their susceptibility to degradation.

Each steam generator contains 3,592 thermally treated alloy 690 tubes with an outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. In addition to the tubing material enhancement, other design enhancements include an increased row 1 U-bend radius and heat-treated low row U-bends. The tube support structure consists of one octofoil flow distribution plate, seven support plates with quatrefoil holes, and six sets of anti-vibration bars. The flow

distribution plate, support plates and anti-vibration bars are made from corrosion-resistant type 405 stainless steel.

The tubesheet is nominally 21 inches thick and is made of ASME SA-508 Class 2a low alloy steel forging material with Inconel cladding on the primary side. All tubes are hydraulically expanded into the tubesheet holes. The expansion includes essentially the entire depth of the tubesheet on both the hot and cold leg ends.

Effective Full Power Years of Operation

The replacement steam generators were placed into service in March 1989 at the beginning of operating cycle 7. Until the cycle 11 shutdown that commenced in September 1997, the replacement steam generators accumulated a runtime of approximately 5.8 effective full power years.

Degradation History

Tube inspections have been conducted on four occasions since the steam generators were replaced in 1988. During these inspections, no corrosion-related degradation has been discovered. A small number of tubes have been removed from service, due to foreign object-induced wear, mechanical damage incurred during offline pressure pulse cleaning operations, and a manufacturing defect.

Given the population of 14,368 tubes, the plugging total of 15 tubes indicates that 0.10% of the tubes have been plugged. This value is well below the overall licensed plugging limit of 10%. The steam generator with the highest plugging level has six tubes plugged, resulting in a peak steam generator plugging level of 0.17%, which is also well below the licensed plugging limit of 15% in any one steam generator.

Based upon the most recent inspection results, the Unit 2 steam generators are considered free from any active degradation mechanisms.

4.0 1997 Inspection Summary

To satisfy T/S, a minimum full-length tube examination consisting of 3% of the in-service tubes in each of the four steam generators was required during the 1997 inspection. The actual inspection scope greatly exceeded these requirements. The base inspection encompassed a full-length bobbin coil inspection of 50% of the tubes in each of the four steam generators. Additionally, a 20% sample hot leg tubesheet examination was performed in one steam generator. The tubesheet examination used a 3-coil rotating coil that contained a 0.115" pancake coil, which contains coils sensitive to both axially and circumferentially oriented flaws.

Inspection results were very favorable with no evidence of corrosion-induced degradation identified and only minor wear indications found in select areas. The wear indications were identified during the bobbin coil examination and affected three tubes in steam generator 21. The following table summarizes the indications.

Row	Column	Support	Depth
6	53	5th Hot	6%
		6th Hot	11%
6	54	4th Hot	7%
		5th Hot	4%
1	70	1st Cold	28%

The wear indications reported in Row 6 were caused by fretting at the lower edge of the tube support. Due to the lack of previous inspection history, a degradation growth rate history could not be established for these indications. Due to the minor nature of the wear, they were left in service. The indication in Row 1/Column 70 was believed to have been caused by a burr or a small foreign object near the center of the quatrefoil support. Due to a lack of previous inspection history, a degradation growth rate history could not be established for this indication. The tube at row 1/column 70 was removed from service by plugging since, based upon an assumed growth rate of one cycle, the indication could exceed the repair limit before the next inspection.

The initial bobbin coil examination in steam generator 21 was expanded to ensure that 100% of the tubes in rows 1-3 were inspected. This expansion and the subsequent results provided further assurance that no indications similar to that in row 1/column 70 were present.

The only remaining significant indications were four tube indications that were reported during the rotating coil examination. The affected tubes were all located in steam generator 24 and contained single volumetric indications at the top of the tubesheet. These were typical of indications caused by a loose part and the associated tubes were ultimately removed from service. The four indications are summarized below.

Row	Column	Location
33	15	Tubesheet Hot +0.49"
33	16	Tubesheet Hot +0.05"
33	17	Tubesheet Hot +0.84"
38	72	Tubesheet Hot +0.07"

Because of the foreign object concern, the initial bobbin coil inspection scope in steam generator 24 was expanded to include 100% of the row 1-3 tubes and all the tubes around the periphery (two rows in). The basis for the expansion in this area was to determine if any tube damage

occurred from a foreign object at its point of entry into the bundle. Examinations with a bobbin coil and a rotating coil were performed in the tubes surrounding the loose part volumetric indications. A subsequent secondary side examination did not find a loose part in the row 33 locations, but did locate and remove a small metal wire from the general area of the row 38 indication. The origin of the foreign object could not be determined.

Therefore, the only indications of interest that remain in service involve two tubes in steam generator 21 with minor wear indications on the hot leg supports. These indications are not structurally significant since even the largest indication is only 11% through-wall, or approximately 28% of the T/S repair limit of 40% through-wall. Assuming that this indication grew to its present size over the previous operating cycle and that a similar growth rate would apply during the future operating cycles, the indication would not exceed the repair limit for more than two additional cycles of operation. Therefore, these indications do not threaten structural or leakage integrity during the current cycle. These tubes will continue to be monitored during future inspections.

Primary side inspection results were acceptable with wear identified as the only degradation mechanism present. Overall, tube wear is not a concern since at the levels observed, it is not considered an active degradation mechanism as defined by the EPRI PWR Steam Generator Examination Guidelines, Revision 5. These guidelines define an active degradation mechanism as follows:

> A combination of ten or more new indications of degradation (> 20% through-wall) and previous indications of degradation which display an average growth rate equal to or greater than 25% of the repair limit per cycle in any one steam generator or, one or more new or previously identified indications of degradation, including cracks, which display a growth greater than or equal to the repair limit in one cycle of operation.

In addition to the eddy current tube inspection, other measures were taken to assure the integrity of each steam generator during the previous inspection. Sludge lancing was performed in each steam generator to remove accumulated sludge deposits. Internals inspections were performed to verify the general conditions in select areas of the tube bundle as well as to determine the effectiveness of the sludge lance operation. Additionally, foreign object search and retrieval operations were conducted, resulting in the location and removal a small piece of weld wire from one steam generator. No degradation was noted during the secondary side inspections.

5.0 Steam Generator Chemistry Control

In May 2000, an outside contractor performed an independent review of the station chemistry program affecting the Unit 2 steam generators. This review included primary and secondary

chemistry, layup conditions, and chemical excursions during the previous operating cycle. I&M reviewed and accepted the contractor's findings, which are summarized below.

5.1 Primary Chemistry (Cycle 11)

Hydrogen and lithium controls are the mainstay of the primary chemistry control measures that affect primary water stress corrosion cracking and therefore the integrity of the steam generator tubes. Hydrogen was maintained within its prescribed limits throughout the cycle. Lithium was maintained below 3.5 ppm for the entire cycle, except for several hours during the first day of the cycle. This period is not considered long-term by EPRI guidelines. Chemical contaminants, including chloride, fluoride, sulfate, and dissolved oxygen were also maintained below limits throughout the operating cycle. Since these parameters were within specification for the entire cycle, except where noted below, they are not perceived to have affected the integrity of the steam generators or their ability to perform their intended function during the subsequent operating cycle. Steam generator eddy current inspection results confirmed the absence of primary water stress corrosion cracking after cycle 11 operation.

5.2 Secondary Chemistry

Hideout return data indicate that fouling is not significant in the Unit 2 steam generators. Feedwater iron data indicate that the mass of iron in the feedwater during the cycle is within the limits of industry guidelines. Copper transport rates are low and therefore are not expected to impact steam generator integrity. Hideout return data also indicate that crevices, if they exist, have a near neutral pH. Steam generator non-destructive examination data indicate that chemistry control during operation did not negatively impact steam generator integrity. No degradation or abnormalities were noted during secondary side inspections. Conclusions from the data evaluation are that chemistry control during cycle 11 and the following shutdown did not create conditions that would affect the integrity of the steam generators or their ability to perform their intended function during the subsequent operating cycle.

Layup Program

Bulk water chemistry was adequately maintained throughout the layup from November 1997 through May 2000 to prevent corrosion of the steam generator secondary side. All samples for chloride, sulfate, sodium, and dissolved oxygen were well within limits. There were brief periods when pH and hydrazine were below their minimum limits. Chemistry outside of specification was corrected.

Due to inspection and repair activities, there were periods during the extended shutdown when the steam generators were partially drained and periods when nitrogen sparging was not available. This is consistent with normal industry practices for planned outage maintenance. From September 1999 to March 2000, the steam generators could not be sampled due to

replacement of the sampling panels. Layup chemistry before and after this period was verified to be acceptable. Representative samples could not be obtained for oxygen, but the hydrazine concentrations indicate that oxygen was not present. The steam generators were again isolated and sampling was suspended between late March and early May 2000. Hydrazine concentration decreased during this time, but the dissolved oxygen concentration was verified to be zero when sampling resumed. In summary, layup chemistry did not create conditions that would adversely affect the integrity of the steam generators or their ability to perform their intended function during the subsequent cycle of operation.

5.3 Past Unit 2 Chemical Excursions

A review was conducted of cycle 11 chemistry to identify and evaluate any significant chemical excursions. The chemistry program establishes action levels, consistent with the industry (EPRI) guidelines, to ensure that remedial actions are taken when monitored parameters are outside the recommended range. During Cycle 11, a total of 154 hours of Action Level 1 time and 16 hours of Action Level 2 time was documented. The majority of this action level time was due to elevated steam generator chloride during unit startup following refueling and mid-cycle elevated condensate dissolved oxygen concentration due to plant testing for condenser lake water in-leakage. No evidence of steam generator degradation from these Action Level 1 and Action Level 2 conditions was detected. EPRI Action Level 3 entries correspond to the presence of conditions that will result in rapid steam generator corrosion. The review confirmed that there were no Action Level 3 chemistry excursions during cycle 11.

5.4 Conclusion

Chemistry control during cycle 11 and the following extended shutdown did not create conditions that would affect the integrity of the steam generators or their ability to perform their intended function during the subsequent cycle of operation.

6.0 Impact of Extended Unit Shutdown

The integrity of primary and secondary side components was verified during the 1997 inspection. Since that time, conditions have been maintained in a wet layup environment (with exceptions noted above). The temperature, pressure, and chemistry conditions that typically initiate or accelerate tubing degradation are absent during wet layup. Therefore, the steam generators are believed capable of completing cycle 12 without requiring a mid-cycle shutdown for inspection. This conclusion is supported by the following factors:

- The Unit 2 steam generators have enhanced material and design features that reduce their susceptibility to degradation. The tubes are constructed of corrosion-resistant alloy 690. Industry experience has demonstrated that alloy 690 has significantly enhanced corrosion resistance over its alloy 600 predecessor.
- No active degradation mechanisms, based on EPRI definitions, are present in any of the four Unit 2 steam generators. Based on past inspection results, they have experienced negligible degradation, demonstrated by an overall plugging percentage of 0.10%.
- The only significant indications that were detected in tubing that remains in service are wear indications of no more than 11% through-wall, which are not influenced when the steam generator is out of service.
- The extended unit shutdown is not expected to have degraded the stainless steel support plates or other steam generator internals. The lack of power operation minimizes service-based degradation such as wear. No degradation concerns are expected due to the layup environment that has been maintained in the steam generators. Chemistry conditions during layup are discussed in section 5 of this attachment.
- Sludge lancing was performed in 1997. Subsequent visual inspections determined the tube sheet area to be free of sludge and verified the overall good conditions of select secondary side areas.
- Throughout the shutdown period, the steam generators were maintained under layup conditions, absent the temperature and pressure conditions typically associated with corrosion-based degradation.
- No chemical excursions that would result in rapid steam generator corrosion occurred during the previous operating cycle, as both primary and secondary chemistry were well maintained.
- As of September 26, 2000, the calculated steam generator leak rate was zero gallons per day after 99 days of Mode 1 operation.

In summary, as evidenced by the expanded inspection programs, the enhanced steam generator design features and chemistry control program have been successful in the prevention and mitigation of steam generator degradation. The extended unit shutdown has not created conditions that would adversely affect the integrity of the steam generators or their ability to operate safely and reliably during the next operating cycle.

7.0 Related Topics

The following is a brief description of additional barriers that are in place to help ensure tube integrity and protect against tube rupture.

Steam Generator Program

CNP has endorsed the NEI Steam Generator Program Initiative such that future inspections are expected to be consistent with NEI 97-06 and the EPRI PWR Steam Generator Examination Guidelines, Revision 5.

Foreign Material Exclusion

CNP has implemented a controlled foreign material exclusion program to preclude the introduction of foreign objects into either the primary or the secondary side of plant systems. Adherence to this program is a requirement for inspection and repair operations conducted on the steam generators.

Digital Metal Impact Monitoring System (DMIMS)

The DMIMS provides early detection of loose parts in the reactor coolant system which may occur as result of material wear, component failure or outage/maintenance work. The system alerts operators to significant impacts from loose parts by an alarm. The DMIMS provides operators with a digital display of the current impact status, printed reports summarizing the past days events, and the amplified sound played through a speaker. Accelerometers are mounted above and below the tube sheet of each steam generator to supply a signal to the DMIMS.

Primary-to-Secondary Leak Rate Monitoring

No detectable primary-to-secondary leakage was recorded during cycle 11. Primary-to-secondary leak rate monitoring at CNP generally follows the EPRI PWR Primary-to-Secondary Leak Guidelines. However, there are some differences due to site-specific conditions. CNP does not employ the requirements for the rate-of-change in leak rate, which is included in the EPRI guidelines. This stems from the lack of defect propagation data on alloy 690 tubing on which to base site-specific action levels and the conservatism of the new administrative limit. An administrative limit for primary-to-secondary leakage provides for a unit shutdown in the event leakage meets or exceeds 75 gallons per day. This value is very conservative with respect to Unit 2 T/S 3.4.6.2c, which limits primary-to-secondary leakage to 500 gallons per day through any one steam generator. As of September 26, 2000, the calculated steam generator leak rate was zero gallons per day after 99 days of Mode 1 operation.

Review of Industry Experience

I&M has reviewed recent industry experience regarding the quality of steam generator inspections and the corresponding tube integrity assessments. Related concerns have surfaced at plants with alloy 600 tubes that have significant degradation problems. These concerns focus primarily on inspection techniques and procedures, specific corrosion mechanisms, and predicted tube integrity.

While the CNP Unit 2 steam generators have thermally treated alloy 690 tubes that do not have concerns with corrosion or cracking, these industry issues were investigated for applicability to this proposed extension.

The inspection techniques used in the 1997 CNP Unit 2 steam generator inspection were recently reviewed by I&M to ensure the reliability of inspection data. The test equipment and techniques used met all applicable industry standards. Signal noise due to deposits on the tubes has been a problem at other plants, but not been a concern at CNP Unit 2.
EPRI has collected inspection data from 35 nuclear plants that have replacement steam generators with thermally treated alloy 690 tubes, including 14 plants in the United States. None of these plants has reported any stress corrosion cracking or other corrosion-related degradation in their replacement steam generators.

The planned scope of the 1997 inspection encompassed a far greater percentage of tubes than required by T/S. In the few areas where tube indications were found, the inspection scope was appropriately expanded. No additional indications were subsequently detected. There were no degraded tubes (20% or more through-wall) left in service. Flaw characterization and predicted growth rates are not a concern at CNP Unit 2 due to the relative lack of degradation.

All analysts involved in the inspection were appropriately trained as EPRI-qualified data analysts. In addition, I&M has arranged for EPRI to perform an independent review of the 1997 Unit 2 steam generator tube inspection techniques and results to confirm the adequacy of the testing and findings.

The material and design enhancements in the Unit 2 steam generators minimize the potential for service-related degradation. This is well supported by their performance history. No concerns regarding the quality of steam generator inspections and the resulting data have been identified. Accordingly, I&M has concluded that the issues reflected in recent steam generator experience are not concerns with CNP Unit 2 steam generators and this proposed extension.

ATTACHMENT 7 TO C0900-04

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this submittal. Other actions discussed in the submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory commission (NRC) for the NRC's information and are not regulatory commitments.

COMMITMENT	DATE
I&M will have the Electric Power Research Institute perform an independent review of the 1997 Cook Nuclear Plant Unit 2 steam generator tube inspection techniques and results. I&M will report any adverse findings from this review to the NRC.	November 30, 2000