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June 12, 1996

10 CFR 50.54(q)

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Emergency Plan Revision 18

Modifications have been made to the Kewaunee Nuclear Power Plant (KNPP) Emergency Plan. A description of the changes and the safety evaluations are included as Attachment 1 to this letter. A copy of the revised Emergency Plan is included as Attachment 2 to this letter. Attachments 3 and 4 are supporting documents for this revision. These changes do not decrease the effectiveness of the plan and continue to meet the standards of 10 CFR 50.47(b) and the requirements of Part 50, Appendix E; therefore, they are made in accordance with 10 CFR 50.54(q) and prior Nuclear Regulatory Commission approval is not required.

Pursuant to 10 CFR 50.4, two additional copies of this letter and attachments are hereby submitted to the U.S. Nuclear Regulatory Commission, Region III. As required, one copy of this letter and attachments is also submitted to the KNPP Senior Resident Inspector.

Sincerely,

M. L. Marchi

M. L. Marchi
Manager - Nuclear Business Group

DRS
Attach.
cc - Mr. Lanny Smith, PSCW w/o attach.
US NRC Senior Resident Inspector w/attach
US NRC, Region III (2 copies) w/attach.

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ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 12, 1996

DESCRIPTION OF CHANGE AND SAFETY EVALUATION

SAFETY EVALUATION
KEWAUNEE NUCLEAR POWER PLANT

EMERGENCY PLAN
REVISION 18

TABLE OF CONTENTS

page i — Description of Change

Updated Senior Vice President - Nuclear Power signature and date on the policy page.

Safety Evaluation

This change provides continuity of management support and does not decrease the effectiveness of the plan.

SECTION 1.0, INTRODUCTION

page 1.2-1 — Description of Change

The Nuclear Emergency Preparedness Coordinator was modified to show one coordinator position.

Safety Evaluation

A company wide personnel reduction policy was imposed when the one of the two coordinator positions was vacated due to a job transfer. The second position was eliminated. Added support was provided from the plant training group and the plant administrative group. This change does not decrease the effectiveness of the plan.

SECTION 4.0, EMERGENCY CONDITIONS, TABLE 4-1 "EMERGENCY CLASSIFICATION"

page
1 of 18

Description of Change

The table showing the association between event type and applicable classification chart shows the deletion of Chart H, " Primary Side Anomaly" and Chart L, "Personal Injury."

Safety Evaluation

The change and evaluation for these charts will be addressed in their respective sections below. This change is administrative in nature and does not decrease the effectiveness of the plan.

Chart A(1)
page 2 of 18

Description of Change

a. In the criteria section for the second site emergency from the top, the phrase "in the environs" was replaced with "at the site boundary." Also the term "whole body" was deleted.

b. In the criteria section for the first alert from the top, the term "Technical Specifications" was replaced with "ODCM," and the information in parentheses was deleted.

c. In the KNPP indication section for the first alert from the top, the reference to liquid releases was deleted.

d. In the KNPP indication section for the second alert form the top, all the radiation monitor values were change to "1.0E+4" and the acronym "MPC" was changed to "DAC values."

e. In the KNPP indication section for the unusual event, the technical specification reference was changed to an ODCM reference. In the criteria section the reference to the "Radiological Effluent Technical Specification" was changed to "Off-site Dose Calculation Manual," and information in the parentheses was deleted.

Safety Evaluation

a. The use of the site boundary reference point is more consistently used in this criteria as compared to other criteria and the removal of the term "whole body" is consistent with current 10 CFR 20 terminology. This change does not decrease the effectiveness of the plan.

b. This change was made for consistency with Kewaunee Technical Specification amendment number 104 which relocated the programmatic control and procedural details for Radiological Effluent Technical Specifications (RETS) from the Kewaunee Technical Specifications to the ODCM. To focus the operators attention to the current reference document (the ODCM), specific values were eliminated from the criteria section. These changes do not decrease the effectiveness of the plan.

c. This reference affirming a negative was not needed. This change does not decrease the effectiveness of the plan.

d. Plant experience has shown that a normal and conservative reading for these area radiation monitors is 10 mR/hr. By applying the 1000 times normal factor the action level is now consistent for user application and is also conservative. The use of DAC values is consistent with current 10 CFR 20 terminology. These changes do not decrease the effectiveness of the plan.

e. This change was made for consistency with Kewaunee Technical Specification amendment number 104 which relocated the programmatic control and procedural details for Radiological Effluent Technical Specifications (RETS)from the Kewaunee Technical Specification to the ODCM. To focus the operators attention to the original reference document (the ODCM), specific values were eliminated from the criteria section. These changes do not decrease the effectiveness of the plan.

Chart A(2)
page 3 of 18 Description of Change

The tables on Revision 15 pages 3 and 4 were consolidated situations of Auxiliary Building vent releases with core damage. No table values were changed. All "notes" were brought to the top of the page above the tables and wording changes were made to improve readability.

Safety Evaluation

These changes provide for a more prompt and efficient use of the procedure and do not decrease the effectiveness of the plan.

Chart A(2)
page 4 of 18 Description of Change

The tables on Revision 15 pages 5 and 6 were consolidated situations of Auxiliary Building vent releases without core damage. No table values were changed. All "notes" were brought to the top of the page above the tables and wording changes were made to improve readability.

Safety Evaluation

These changes provide for a more prompt and efficient use to the procedure and do not decrease the effectiveness of the plan.

Chart A(2)
page 5 of 18 Description of Change

The format of the Steam Line release and Shield Building Stack release were modified for consistency with the previous tables. No change was made to the table values.

Safety Evaluation

These changes provide for a more prompt and efficient use to the procedure and do not decrease the effectiveness of the plan.

Chart B
page 6 of 18 Description of change

In the KNPP indication section for the first unusual event from the top, the indications were rewritten to more closely match the wording of Technical Specifications 3.1.c.

Safety Evaluation

This change improves the accuracy of determination and does not decrease the effectiveness of the plan.

Chart C
page 7 of 18 Description of Change

a. The note at the top of the page was rewritten to provide a clearer statement on when this chart should not be applied (i.e. during steam generator tube rupture situations) and added references to appropriate charts if this situation exists.

b. In the KNPP indicator section for the general emergency, indication 2d, "Subcooling meter is zero" was deleted.

Safety Evaluation

a. This change provides clearer direction for the user and does not decrease the effectiveness of the plan.

b. Zero subcooling is a condition experienced during a large loss of coolant accident and not an indicator of emergency core cooling system failure. This change does not decrease the effectiveness of the plan.

Chart D
page 8 of 18 Description of Change

In the KNPP indication section for unusual event, the primary to secondary leak rate value was changed from "> 500 gpd" to "> 150 gpd."

Safety Evaluation

This change was made to comply with Technical Specification amendment number 118. This change does not decrease the effectiveness of the plan.

Chart E
page 9 of 18 Description of Change

a. In the KNPP indication section for general emergency, a minor format change was made without change to the content.

b. In the KNPP indication section for the second alert from the top, The "loss of off-site and on-site AC power for <15 minutes" was re written. The declaration of an alert should be initiated only if the diesel generators do not respond as designed.

c. In the KNPP indications section for the first unusual event from the top, "or" statement "b" was rewritten to state, "Both D/Gs unavailable (D/Gs unable to supply bus 5 or 6 by any means).

d. In the KNPP indication section for the second unusual event from the top, a minor formatting change was made without change to the content.

Safety Evaluation

a. This change does not decrease the effectiveness of the plan.

b. Although the diesel generators are off when off-site power is lost, the normal start up sequence of the diesel generators would maintain a safe operation condition at the plant and an alert is not warranted. Failure of the diesels to respond would justify declaration of an alert. This change does not decrease the effectiveness of the plan.

c. Per NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, the loss of an ESF function (i.e., Technical Specification in operability) is not by itself an indication of an emergency. If Technical Specification limits requiring shutdown, including standard shutdown sequence, and the required shutdown action can be met within the required time limits, emergency declaration is not warranted. This change does not decrease the effectiveness of the plan.

d. This change does not decrease the effectiveness of the plan.

Chart F
page 10 of 18

Description of Change

a. In the criteria section for the site emergency, background information was deleted.

b. In the KNPP indication section for the first alert from the top, a minor format change was made without change to the content.

c. The unusual event classification for emergency core cooling indicated and discharged to the reactor vessel was deleted.

d. The remaining unusual event classification criteria was changed to read "Inability to reach required shutdown within Tech. Spec. limits. The KNPP indication section was also modified to reflect the change in criteria.

e. In the KNPP indication section for the remaining unusual event classification, the "note" concerning loss of Auxiliary Feed Water was modified to stress the point that an unusual event should be declared if plant procedures for loss of Auxiliary Feed Water are implemented regardless of Technical Specification actions taken.

Safety Evaluation

a. Un-necessary background information detracted from the readability of the procedure. This change does not decrease the effectiveness of the plan.

b. This change does not decease the effectiveness of the plan.

c. Per NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, emergency core cooling system discharge may not warrant a declaration of an unusual event in the context of engineered safety feature anomalies. Conditions requiring emergency core cooling system are adequately covered in Charts C, D, and I. This change does not decrease the effectiveness of the plan.

d. Per NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, the loss of an ESF function (i.e., Technical Specification in operability) is not by itself an indication of an emergency. If Technical Specification limits requiring shutdown, including standard shutdown sequence, and the required shutdown action can be met within the required time limits, emergency declaration is not warranted. This change does not decrease the effectiveness of the plan.

e. A complete loss of this engineered safeguards function should be declared an unusual event, if not superseded by another event classification. This change does not decrease the effectiveness of the plan.

Chart G
page 11 of 18

Description of Change

a. The KNPP indication section for the site emergency and alert were clarified and minor format changes were made without change to the content.

b. The KNPP indication section for unusual event was modified to state that although a significant loss of engineered safeguards function or reactor protection instrumentation does warrant the declaration of an unusual event, a back down prompted by Technical Specifications while the affected parameter remains monitorable does not.

Safety Evaluation

a. This change does not decrease the effectiveness of the plan.

b. Consistent with NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, a Technical Specification back down due to channel failures is not by itself an emergency condition and may not be a significant loss of indications or assessment capability. This change does not decease the effectiveness of the plan.

Chart H
page 11 of 18

Description of Change

This chart was deleted.

Safety Evaluation

Consistent with NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, fuel damage situation are adequately covered in Chart B of this procedure and abnormal coolant temperature and/or pressure situations are adequately covered by following Technical Specification limits for back down. This change does not decrease the effectiveness of the plan.

Chart I
page 12 of 18

Description of Change

a. The KNPP indication and criteria sections for the site emergency were clarified and minor format changes were made without change to the content.

b. In the KNPP indication section for the alert, the reference to "verified by SP 36-082" was deleted along with minor editorial changes that did not change the content of this section.

Safety Evaluation

a. This change does not decrease the effectiveness of the plan.

b. For steam line breaks or events requiring back down due to primary to secondary leakage, Reactor Coolant System leak rate testing (SP 36-082) is not practical because the accuracy to this test depends on plant system stability. Since the Reactor Coolant System would not be in the stable condition, the conservative approach would be to declare the unusual event based on the indications currently listed and not wait for test verification. This change does not decrease the effectiveness of the plan.

Chart J
page 13 of 18

Description of Change

a. In the KNPP indication section for general emergency minor format changes were made without change to the content.

b. The unusual event for loss of containment integrity requiring shut down by Technical Specifications was deleted.

Safety Evaluation

a. This change does not decrease the effectiveness of the plan.

b. Consistent with NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, this situation is not by it self warrant the declaration of an unusual event. A situation of this nature is adequately covered in Chart F. This change does not decrease the effectiveness of the plan.

Chart L
page 14 of 18

Description of Change

This chart was deleted.

Safety Evaluation

Consistent with NRC letter of 7/11/94 (EPPOS No. 1) on acceptable deviation to NUREG-0654, a situation in which an individual is injured and contaminated is not, in itself, indicative of reactor or public safety situation. This change does not decrease the effectiveness of the plan.

Chart M
page 15 of 18

Description of Change

In the KNPP indication section for Unusual Event the "and" was changed to an "or" and an "*" was added at the end of item "b" to direct the reader to a footnote. That footnote directs the reader to contact the U of W Seismic Center for verification.

Safety Evaluation

This change brings the KNPP indication section into closer conformity with the criteria section and provides clearer direction to the reader. This change does not decrease the effectiveness of the plan.

SECTION 5.0, ORGANIZATIONAL CONTROL OF EMERGENCIES

page 5.1-1

Description of Change

In the second paragraph. the position "nuclear computer support" reporting to the Manager - Nuclear Plant Support Services was deleted and replaced with "administrative support."

Safety Evaluation

All computer support was consolidated under one organization for the company. Although computer support staff still report and work at the plant they bave a different reporting chain. Administrative support was added to the groups reporting to the Manager - Nuclear Plant Support Services. This change does not decrease the effectiveness of the plan.

Fig. 5-1.2 Description of Change

a. The Nuclear Emergency Preparedness Coordinator was modified to show one coordinator position.

b. The Nuclear Personnel Supervisor title was changed to Plant Personnel/Budget Coordinator.

c. The Plant Office Supervisor position was added.

Safety Evaluation

a. A company wide personnel reduction policy was imposed when the one of the two coordinator positions was vacated do to a job transfer. The second position was eliminated. Added support was provided from the plant training group and the plant administrative group. This change does not decrease the effectiveness of the plan.

b. Due to a consolidation of positions the responsibilities of personnel and budgeting was also consolidated under one person. This change does not decrease the effectiveness of the plan.

c. This change does not decrease the effectiveness of the plan.

Fig. 5-1.3 Description of Change

The title "Superintendent - Plant Instrument and Control" was changed to "Plant Instrument and Control Supervisor." The position "Group Leader - Instrument and Control Engineering" was added to the figure.

Safety Evaluation

This change is administrative in nature that does not affect the availability of instrument and control support. this change does not decrease the effectiveness of the plan.

Fig. 5-1.5 Description of Change

a. The positions "Special Projects Process Leader" and "Records Management Group Leader" were added to the figure.

b. The title "Projects Evaluation Process Leader" was changed to "Evaluation and Projects Process Leader."

Safety Evaluation

a&b. These changes are administrative in nature and do not decrease the availability of plant support. This change does not decrease the effectiveness of the plan.

SECTION 6.0, EMERGENCY MEASURES

Fig. 6-2 Description of Change

Removed the notification locations of "Wisconsin State Patrol Fond du lac" and "East Central Area EOC."

Safety Evaluation

These are no longer notified directly by the utility and provide no direct or immediate support during the early stages of a declared emergency. They are, however, contacted for support as needed by the State of Wisconsin, Division of Emergency Government. This change does not decrease the effectiveness of the plan.

SECTION 7.0, EQUIPMENT AND FACILITIES

page 7.1-2 Description of Change

In the first paragraph, the phrase "and the National Warning System (NAWAS) Telephone" was deleted.

Safety Evaluation

The NAWAS was removed in total from all utility emergency response facilities in August of 1995. The NAWAS has been for several years a back up to a shared primary notification system call Dial-Select, a dedicated open circuit system. When the Federal Government announced that Federal funding for the system was being eliminated the State and County emergency governments started discussions to eliminate the use of this system also. Since all parties were confident with the operation of the Dial-Select system, the State and Counties were supportive of elimination of the NAWAS at the utilities. This change does not decrease the effectiveness of the plan.

Page 7.2-2 Description of Change

The NAWAS system description was deleted.

Safety Evaluation

The NAWAS was removed in total from all utility emergency response facilities in August of 1995. The NAWAS has been for several years a back up to a shared primary notification system call Dial-Select, a dedicated open circuit system. When the Federal Government announced that Federal funding for the system was being eliminated the State and County emergency governments started discussions to eliminate the use of this system also. Since all parties were confident with the operation of the Dial-Select system, the State and Counties were supportive of elimination of the NAWAS at the utilities. This change does not decrease the effectiveness of the plan.

Fig. 7-2 Description of Change

Deleted the lines connecting facilities by NAWAS.

Safety Evaluation

The NAWAS was removed in total from all utility emergency response facilities in August of 1995. The NAWAS has been for several years a back up to a shared primary notification system call Dial-Select, a dedicated open circuit system. When the Federal Government announced that Federal funding for the system was being eliminated the State and County emergency governments started discussions to eliminate the use of this system also. Since all parties were confident with the operation of the Dial-Select system, the State and Counties were supportive of elimination of the NAWAS at the utilities. This change does not decrease the effectiveness of the plan.

SECTION 8.0, MAINTAINING EMERGENCY PREPAREDNESS

page 8.1-3 Description of Change

In the first paragraph, the title "Manager - Nuclear Engineering" was replaced by the title "Nuclear Communications Coordinator."

Safety Evaluation

The position of Manager - Nuclear Engineering has been changed to Manager - Nuclear Business Group a position that does not have a direct supporting role for the emergency preparedness program. The Nuclear Communications Coordinator does have a direct supporting function and should be added. This change does not decrease the effectiveness of the plan.

APPENDICES

Appx. C,
Fig. C-4
page C-5

Description of Change

The population distribution by evacuation area map was replaced with only minor format changes and no change to content.

Safety Evaluation

This change provides a map that is clearer and easier to read. This change does not decrease the effectiveness of the plan.

Appx. D

Description of Change

All letters of agreement that have been updated since the last revision to this plan were corrected to show the correct and current issue dates.

Safety Evaluation

This change does not decrease the effectiveness of the plan.

Appx. E
page E-1

Description of Change

In the parentheses at the bottom of the text, the word "Coordinator" was replaced with "Supervisor."

Safety Evaluation

This change does not decrease the effectiveness of the plan.

Appx. F Description of Change

Through out this appendix minor modifications were made to resource titles or equipment descriptions. The exception being page F-7, "Site Boundary Facility." The traffic control equipment was eliminated from this list.

Safety Evaluation

This title and equipment description changes provide a more accurate listing of available resources. The use of traffic control equipment such as barricades and flashers have been replaced with the use of security vehicles. These changes do not decrease the effectiveness of the plan.

Appx. G Description of Change

Where appropriate, emergency plan implementing procedure titles were corrected or deleted.

Safety Evaluation

These changes provide an accurate listing of the emergency plan implementing procedures currently in place. These changes do not decrease the effectiveness of the plan.

ATTACHMENT 2

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 12, 1996

<u>EMERGENCY PLAN (REVISED PAGES)</u>

Kewaunee Nuclear Power Plant

June 11, 1996

Revision 18 of the KNPP Emergency Plan

WMBartelme (95)	DRSeebart (15)	STF (98)
MEMowrer (48)	CHutter (36)	STF (76,91,94,97, 99,100,101,102,103, 104,105,106)
DTBraun (83)	JLMueller (52)	KHWeinhauer (30)
DEDay (25)	Maint. Library (43)	QP Lib (82)
JJHannon (44)	KLipp (17,18,51)	Originals - KNPP QA Vault
KKMalley (54)	MTReinhart (55,57)	
AProkash (56)	AIProkash (35)	
CASternitzky (33)		

Enclosed is Revision 18 of the KNPP Emergency Plan. Please follow the directions on the following page.

15 - DRSeebart
17,18 - KLipp (Nuc. Lib.)
25 - DEDay
30 - KHWeinhauer
33 - CASternitzky
35 - AIProkash
36 - CHutter
43 - Maintenance Lib
44 - JJHannon (I&C Shop)
48 - MEMowrer (Security Bldg.)
51 - KLipp (EOF)
52 - JLMueller (TSC)
54 - KKMalley (OSF)
55 - MTReinhart (RAF)
56 - AProkash (SBF)
57 - MTReinhart (RPO)
76 - PJWiese (STF Lib)
82 - QP Lib
83 - DTBraun (SS)
91,94 - PJWiese (STF)
95 - WMBartelme
97 - PJWiese (Off-Site)
98 - PJWiese (ATF-3)
99,100,101,102,103,104,105,106,
- PJWiese (Licensing Requal)

Please note the affected pages for text changes listed on page 5 of the Record of Revisions Section.

Thanks for your help in effecting this revision.

I CERTIFY that this manual has been updated.
SIGNATURE/DATE
When update is complete, please return this page for a record of revision to ***FRAN ARNO - ATF-2***

Dave Seebart/cjq

Dave Seebart/cjq

Enclosure

KEWAUNEE NUCLEAR POWER PLANT EMERGENCY PLAN

RECORD OF REVISIONS

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KEWAUNEE NUCLEAR POWER PLANT EMERGENCY PLAN

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KEWAUNEE NUCLEAR POWER PLANT EMERGENCY PLAN

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KEWAUNEE NUCLEAR POWER PLANT EMERGENCY PLAN

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POLICY STATEMENT - EMERGENCY PLAN

Wisconsin Public Service Corporation is fully committed to the establishment and maintenance of an effective emergency preparedness program. This program will not only encompass the Emergency Plan itself, but also the procedures, facilities, equipment and training needed to accomplish the standards set forth in the Emergency Plan. All levels of management have a strong commitment to emergency preparedness, and each employee must take responsibility for actions necessary to implement a successful emergency preparedness program.

Wisconsin Public Service Corporation recognizes the fact that at times there will be differences between portions of this Emergency Plan and the Emergency Plan Implementing Procedures (EPIPs). The Emergency Plan is the guiding document to which the procedures are written, and as such it describes the organization, emergency measures, training, etc., of WPSC's emergency preparedness program in general terms. The EPIPs more accurately reflect the actual approach to how a particular situation will be addressed, specific assignment of personnel, placement of equipment, etc.

As long as the differences between specific implementing procedures are not substantive and the intent or commitment of the plan is not compromised, the procedure will reflect the actual plant activity or commitment.

Approved By Clark Steinhardt / 5/22/96

Clark R. Steinhardt, Senior Vice President - Nuclear Power/Date

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In addition to the Emergency Plan, detailed Emergency Plan Implementing Procedures (EPIPs) and a Nuclear Emergency Public Information Plan and Procedures have been developed and are available for use at the WPSC emergency response facilities. Nuclear Administrative Directives and Emergency Plan Maintenance Procedures have been developed for maintaining the emergency preparedness program and are available in the plant and corporate nuclear libraries.

A cross-reference between the Emergency Plan sections and corresponding EPIPs appears in Appendix G of this Emergency Plan.

1.2 RESPONSIBILITIES WITH RESPECT TO MAINTAINING EMERGENCY PREPAREDNESS

As the licensed operator of a nuclear power reactor under 10 CFR Part 50, Wisconsin Public Service Corporation (WPSC) has the primary responsibility for the planning and implementation of emergency measures within the site boundaries of the Kewaunee Nuclear Power Plant. The Senior Vice President - Nuclear Power has the overall authority and responsibility for radiological emergency response planning, to assure that an adequate level of emergency preparedness is established and maintained by WPSC in support of the Kewaunee Nuclear Power Plant. The Manager - Nuclear Plant Support Services supported on-site by a Nuclear Emergency Preparedness Supervisor and a coordinator, is responsible for assuring that adequate nuclear power production support is provided to the emergency preparedness program.

Wisconsin Public Service Corporation recognizes that advance agreements with Federal, State, and local organizations are necessary to obtain additional emergency support services and equipment. The agencies with which WPSC has agreements are listed in Appendix D of this plan and the letters of agreement are kept on file by WPSC. Wisconsin Public Service Corporation coordinates its efforts with Federal, State, and local organizations in planning emergency response activities and operations.

projected doses within the site boundary and off-site areas.

4.2.2 Classification of Postulated Accideuts

The events postulated in Section 14 of the USAR may be categorized into one or more of the four emergency classifications. TABLE 4-2 lists each of these design basis events and the emergency classifications that most likely relate to the event according to the classification criteria discussed in Section 4.1.

TABLE 4-1

EMERGENCY ACTION LEVEL CHARTS

The following charts are separated into different abnormal operating conditions which may, depending upon their severity, be classified as an Unusual Event, Alert, Site Emergency, or General Emergency.

	CHART	PAGE
Abnormal Radiological Effluent	A (1)	2
Gaseous Effluent Action Levels	A (2)	3 - 5
Fuel Damage Indication	B	6
Primary Leak to LOCA	C	7
Primary to Secondary Leak	D	8
Loss of Power	E	9
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Loss of Indication	G	11
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Secondary Side Anomaly	I	12
Miscellaneous Abnormal Plant Conditions	J	13
Fire and Fire Protection	K	14
DELETED	L	14
Earthquake	M	15
High Winds or Tornado	N	15
Flood, Low Water, or Seiche	O	16
External Events and Chemical Spills	P	17
Security Contingency	Q	18

TABLE 4-1 CHART A(1)
ABNORMAL RADIOLOGICAL EFFLUENT

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
SEE CHART A(2)	Effluent monitors detect levels corresponding to greater than 1 rem/hr whole body or 5 rem/hr thyroid at the site boundary under "actual meteorological" conditions.	GENERAL EMERGENCY
Projected or measured dose rates to be provided by the Radiological Protection Director or Environmental Monitoring Teams.	Projected or measured in the environs dose rates greater than 1 rem/hr whole body or 5 rem/hr thyroid at the site boundary.	GENERAL EMERGENCY
SEE CHART A(2)	Effluent monitors detect levels corresponding to greater than 50 mr/hr for ½ hour OR greater than 500 mr/hr for two minutes (or five times these levels to the thyroid) OR for "adverse meteorology."	SITE EMERGENCY
Projected or measured dose rates to be provided by the Radiological Protection Director or Environmental Monitoring Teams.	At the site boundary, projected or measured dose rates greater than 50 mr/hr for ½ hours OR greater than 500 mr/hr for two minutes (or five times these levels to the thyroid) or EPA PAGs are projected to be exceeded outside the site boundary.	SITE EMERGENCY
SEE CHART A(2)	Radiological effluents greater than 10 times ODCM instantaneous limits.	ALERT
a. Containment R-2 ≥ 1.0E+4 mr/hr OR b. Charging Area R-4 ≥ 1.0E+4 mr/hr OR c. SFP Area R-5 ≥ 1.0E+4 mr/hr OR d. Plant area air sample indicates airborne contamination > 1000 times the occupational DAC values.	Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials (e.g., radiation levels suddenly increase by a factor of 1000).	ALERT
(1) Gaseous Releases: See Chart A(2) (2) Liquid Releases: Notification by the Rad-Chem Group of violating ODCM 3.3.1 limits.	Off-site Dose Calculation Manual limits exceeded.	UNUSUAL EVENT

TABLE 4-1 CHART A(2)
GASEOUS EFFLUENT ACTION LEVELS

I. AUX BUILDING VENT RELEASES - *WITH SIGNIFICANT CORE DAMAGE*

Instrument readings assuming a post-accident gas release and Significant Core Damage (Containment High Range Radiation Monitors 42599 (R-40) and 42600 (R-41) reads 1000 R/hr within one-half hour of the accident.

NOTE: Use adverse meteorology conditions (ADV MET) only when, 10m and 60m wind speed < 5mph AND Delta-T >2.4 degrees F. All other cases are average meteorology (AVG MET).

NOTE: R-13 and R-14 are expected to be off scale high during all events on this page.

SV & SFP FANS	AUX BLDG SPING MONITORS				AUX BLDG STACK MONITORS				EMERG. CLASS.
TOTAL NUMBER RUNNING	MID RANGE CPM (01-07) PPCS PT G9086G		HIGH RANGE CPM (01-09) PPCS PT G9088G		R-35 MR/HR		R-36 R/HR		
	AVG MET	ADV MET	AVG MET	ADV MET	AVG MET	ADV MET	AVG MET	ADV MET	
1	**	1.1E+4	6.5E+1	*	**	7.9E+2	1.27E+2	7.9E-1	GENERAL EMERG.
2	8.8E+5	5.5E+3	3.25E+1	*	**	3.9E+2	6.35E+1	4.0E-1	
3	5.9E+5	3.7E+3	2.16E+1	*	**	2.6E+2	4.2E+1	2.6E-1	
4	4.4E+5	2.7E+3	1.62E+1	*	**	2.0E+2	3.175E+1	2.0E-1	
1	8.8E+4	5.5E+2	3.0E+0	*	6.3E+3	3.9E+1	6.3E+0	*	SITE EMERG.
2	4.4E+4	2.7E+2	1.5E+0	*	3.1E+3	1.9E+1	3.1E+0	*	
3	2.9E+4	1.8E+2	1.0E+0	*	2.1E+3	1.3E+1	2.1E+0	*	
4	2.2E+4	1.3E+2	*	*	1.5E+3	9.5E+0	1.5E+0	*	
1	1.0E+3	6.2E+0	*	*	7.0E+1	*	*	*	ALERT
2	5.0E+2	3.1E+0	*	*	3.5E+1	*	*	*	
3	3.3E+2	2.0E+0	*	*	2.3E+1	*	*	*	
4	2.5E+2	1.5E+0	*	*	17.5E+1	*	*	*	
1	1.0E+2	6.2E-1	*	*	7.0E+0	*	*	*	UNUSUAL EVENT
2	5.0E+1	3.1E-1	*	*	3.5E+0	*	*	*	
3	3.3E+1	2.0E-1	*	*	2.3E+0	*	*	*	
4	2.5E+1	1.5E-1	*	*	1.7E+0	*	*	*	

* Offscale Low ** Offscale High (Confirmation Only)

2. AUX BUILDING VENT RELEASES - ***WITHOUT CORE DAMAGE***

<u>NOTE</u>: Use adverse meteorology conditions (ADV MET) only when, 10m and 60m wind speed < 5mph <u>AND</u> Delta-T >2.4 degrees F. All other cases are average meteorology (AVG MET).

<u>NOTE</u>: R-13 and R-14 are expected to be off scale high during all events on this page.

SV & SFP FANS	AUX BLDG SPING MONITORS				EMERG. CLASS.
TOTAL NUMBER RUNNING	MID RANGE CPM (01-07) PPCS PT G9086G		HIGH RANGE CPM (01-09) PPCS PT G9088G		
	AVG MET	ADV MET	AVG MET	ADV MET	
1	**	9.4E+4	1.6E+4	1.0E+2	GENERAL EMERG.
2	**	4.7E+4	8.0E+3	5.0E+1	
3	**	3.1E+4	5.3E+3	3.3E+1	
4	**	2.3E+4	4.0E+3	2.5+1	
1	7.5E+5	4.6E+3	8.0E+2	5.0E+0	SITE EMERG.
2	3.7E+5	2.3E+3	4.0E+2	2.5E+0	
3	2.5E+5	1.5+3	2.6E+2	1.6E+0	
4	1.8E+5	1.1E+3	2.0E+2	1.2E+0	

SV & SFP FANS TOTAL NUMBER RUNNING	AUX BLDG SPING MONITORS		EMERG. CLASS.
	LOW RANGE µCi/cc (01-05) PPCS PT G9086G	MID RANGE CPM (01-07) PPCS PT 9086G	
1	**	8.6E+3	ALERT
2	**	4.3E+3	
3	**	2.8E+3	
4	**	2.1E+3	
1	6.3E-2	8.6E+2	UNUSUAL EVENT
2	3.1E-2	4.3E+2	
3	2.1E-2	2.8E+2	
4	1.5E-2	2.1E+2	

** Offscale High (Confirmation Only)

3. **STEAM LINE RELEASE _WITH SIGNIFICANT CORE DAMAGE_**

Instrument readings assuming radioactive steam is releasing at a total of 1.4E+5 pounds per hour to the atmosphere and significant core damage (containment high range radiation monitor 42599 (R-40) or 42600 (R-41) reads 1000 R/hr within one-half hour of the accident).

	"A" Steam Line Monitors		"B" Steam Line Monitors		Emergency Classification
R-15 (cpm)	R-31 (mR/hr)	R-32 (R/hr)	R-33 (mR/hr)	R-34 (R/hr)	
**	1.3E+3	E+0	1.3E+03	E+0	General Emergency
**	6.0E+1	–	6.0E+1	–	Site Emergency
**	1.5E-1	–	1.5E-1	–	Alert
2.0E+05	–	–	–	–	Unusual Event

** Offscale High (Confirmation Only)

4. **SHIELD BUILDING STACK RELEASE**

Instrument readings assuming SBV System is operating in the recirculation mode.

Reactor Bldg. Discharge Vent SPING		Emergency Classification
PPCS PT G9077G (02-07) Mid Range (cpm)	PPCS PT G9079G (02-09) High Range (cpm)	
1.3E+05	1.5E+2	General Emergency
6.7E+03	7.0E+0	Site Emergency
1.5E+1	–	Alert
–	–	Unusual Event

TABLE 4-1 CHART B
FUEL DAMAGE INDICATION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
Any core melt situation with large fission product releases from containment possible or major fuel failure.	Plant conditions exist that make the release of large amounts of radioactivity in a short time period possible.	GENERAL EMERGENCY
(Applies when more than one spent fuel element is damaged.) (1) Fuel Handling Accident in Containment Report of a large object dropped in Rx core or dropped spent fuel assembly **AND** Alarm on R-11 or R-12 (2) Fuel Handling Accident in Auxiliary Bldg. Report of a large object dropped in spent fuel pool dropped spent fuel assembly or loss of water level below spent fuel, **AND** Alarm on R-13 or R-14.	Major damage to spent fuel in containment or auxiliary building.	SITE EMERGENCY
R-9 indication is offscale high **AND** Laboratory analysis confirms RCS activity levels comparable to USAR Table D.4-1.	Severe loss of fuel cladding a. Very high coolant activity sample b. Failed fuel monitor indicates greater than 1% fuel failures within 30 minutes or 5% total fuel failures.	ALERT
(1) Fuel Handling Accident in Containment A confirming report **AND** Alarm on R-11 or R-12 (2) Fuel Handling Accident in Auxiliary Bldg. A confirming report **AND** Alarm on R-13 or R-14.	Fuel damage accident with release of radioactivity to containment or auxiliary building.	ALERT
With RCS Temperature > 500°F, a. >1.0 μCi/gram DOSE Equivalent I-131 for 48 hours, **OR** b. Exceeding T.S. figure 3.1-3 for Dose Equivalent I-131, **OR** c. $> 91/\bar{E}$ μCi/cc As determined by SP 37-065 (from T.S. 3.1.c)	High reactor coolant activity sample.	UNUSUAL EVENT
R-9 is greater than 5.0 R/hr **AND** Verified by RCS chemistry sample analysis.	Failed fuel monitor indicates greater than 0.1% equivalent fuel failures within 30 minutes.	UNUSUAL EVENT

TABLE 4-1 CHART C
PRIMARY LEAK TO LOCA

NOTE: This chart does not apply when leakage from the Reactor Coolant System is caused by a Steam Generator tube rupture.

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) LOCA is verified per IPEOP E-1 "Loss of Reactor or Secondary Coolant" **AND** (2) ECCS failure is indicated by: a. SI and RHR pumps not running **OR** b. Verification of no flow to the reactor vessel **OR** c. Core exit thermocouples indicate greater than 1800°F **AND** (3) Failure or potential failure of containment is indicated by: a. Physical evidence of containment structure damage **OR** b. Loss of all containment fan coil units and both trains of ICS **OR** c. Containment hydrogen monitor indicates ≥10% hydrogen concentration **OR** d. Containment pressure exceeds 46 psig.	(1) Loss of coolant accident **AND** (2) Initial or subsequent failure of ECCS, **AND** (3) Containment failure or potential failure exists (loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier).	GENERAL EMERGENCY
SI System is activated and RCS leakage exceeds charging system capacity as verified by Control Room indications or IPEOPs.	Reactor Coolant System leakage greater than make-up pump capacity.	SITE EMERGENCY
Charging flow versus let down flow indicates leakage >50 GPM from an unidentified source.	Reactor Coolant System leak rate greater than 50 GPM.	ALERT
Initiation of reactor shutdown required by Technical Specification, Section T.S. 3.1.d. Indicated leakage may be determined nsing Reactor Coolant System mass balance calculations performed by SP-36-082.	Exceeding Reactor Coolant System leak rate, Technical Specifications, requiring reactor shutdown.	UNUSUAL EVENT

TABLE 4-1 CHART D
PRIMARY TO SECONDARY LEAK

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Entry into IPEOP E-3 "Steam Generator Tube Rupture" is expected or has occurred **AND** (2) Primary to secondary flow >800 GPM or RCS pressure decreasing uncontrollably **AND** (3) All three transformers Main Aux., Reserve Aux., and Tertiary Aux., are de-energized.	Rapid failure of steam generator tubes with loss of off-site power.	SITE EMERGENCY
(1) Entry into IPEOP E-3 "Steam Generator Tube Rupture" is expected or has occurred **AND** (2) Primary to secondary leak rate >400 GPM **AND** (3) All three transformers: Main Aux., Reserve Aux., and Tertiary Aux., are de-energized.	Rapid gross failure of one steam generator tube with loss of off-site power.	ALERT
(1) Entry into IPEOP E-3 "Steam Generator Tube Rupture" is expected or has occurred **AND** (2) Primary-to-secondary leak rate greater than 800 GPM indicated by SI flow or RWST level change.	Rapid failure of multiple steam generator tubes.	ALERT
Primary to secondary leakage >150 gallons per day for more than 4 hours (TS 3.1.d.2). **Do not delay declaration** if leakage suddenly increases above 150 gallons per day and plant shutdown actions are initiated.	Exceeding Primary to Secondary leak rate Technical Specification.	UNUSUAL EVENT

TABLE 4-1 CHART E
LOSS OF POWER

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
RCS is ≥350°F (1) Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6 **AND** (2) Loss of the turbine driven AFW pump **AND** (3) Conditions exist for greater than 2 hours.	Failure of off-site and on-site AC power **AND** Total loss of auxiliary feedwater makeup capability for greater than 2 hours. (Loss of power plus loss of all AFW would lead to clad failure and potential containment failure.)	GENERAL EMERGENCY
Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6 for longer than 15 minutes. (Does not apply when core is unloaded or cavity is flooded with internals removed.)	Loss of off-site power **AND** Loss of on-site AC power (for more than 15 minutes).	SITE EMERGENCY
Low voltage lockout or de-energized condition on all safeguards DC distribution cabinets for greater than 15 minutes. a. BRA 102 and BRB 102 **OR** b. BRA 104 and BRB 104 (Does not apply when core is unloaded or cavity is flooded with internals removed.)	Loss of all vital on-site DC power (for more than 15 minutes).	SITE EMERGENCY
Low voltage lockout or de-energized condition on all safeguards DC distribution cabinets for less than 15 minutes. a. BRA 102 and BRB 102 **OR** b. BRA 104 and BRB 104 (Does not apply when core is unloaded or cavity is flooded with internals removed.)	Loss of all vital on-site DC power (for less than 15 minutes).	ALERT
Buses 1 through 6 are de-energized, **AND** the D/G supplies to buses 5 and 6 do not respond as designed. AC power is restored to bus 5 or 6 within 15 minutes. (Does not apply when core is unloaded or cavity is flooded with internals removed.)	Loss of off-site power **AND** Loss of on-site AC power (for less than 15 minutes.)	ALERT
With the Reactor Coolant System above cold shutdown condition: a. All three transformers: Main Aux., Reserve Aux., and Tertiary are de-energized **OR** b. Both D/Gs unavailable (unable to supply bus 5 or 6 by any means).	Loss of off-site power **OR** Loss of on-site power capability.	UNUSUAL EVENT
Core is unloaded or reactor cavity is flooded with internals removed **AND** Buses 1 through 6 are de-energized including the D/G supplies to buses 5 and 6 for longer than 15 minutes.	Loss of off-site power **AND** Loss of on-site AC power (for more than 15 minutes).	UNUSUAL EVENT

TABLE 4-1 CHART F
ENGINEERED SAFETY FEATURE ANOMALY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
RCS ≥350°F with a loss of cooling capability or inventory control: a. Loss of negative reactivity control **OR** b. Steam dump, S/G safeties, and power operating reliefs not operable **OR** c. Inability to feed S/Gs at HSD conditions (No AFW or Main Feedwater Flow) **OR** d. Loss of RCS inventory control. A Site Emergency should be declared upon the initiation of bleed and feed per FR H.1, "Response to Loss of Secondary Heat Sink"	Complete loss of any function needed for plant hot shutdown.	SITE EMERGENCY
(Apply this criteria when the RCS is <350°F.) (1) Loss of both trains of RHR **AND** (2) The inability to sustain either natural or forced circulation with the steam generators. (Does not apply when core is unloaded or cavity is flooded with internals removed.)	Complete loss of any function required for cold shutdown.	ALERT
Failure of both Rx trip breakers to open upon receipt of a valid signal. Applies even if IPEOP FR S.1 is not entered.	Failure of the Reactor Protection System to initiate and complete a reactor trip which brings the reactor subcritical.	ALERT
(1) Loss of ESF function, required support function or required Tech Spec instruments OR Exceeding Tech Spec Safety Limits **AND** (2) upon discovery, inability or failure to take required shutdown or mode change actions within the required time. <u>NOTE</u>: Total loss of AFW system when required (FR-H.1 implemented) should be declared a UE regardless of Tech Spec action compliance.	Inability to reach required shutdown within Tech Spec limits	UNUSUAL EVENT

TABLE 4-1 CHART G
LOSS OF INDICATION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Total loss of Annunciator System computer alarms, and sequence of events-recorder for greater than 15 minutes **AND** (2) Uncontrolled plant transient in progress or initiated during the loss.	Most or all alarms (annunciators) lost and a plant transient initiated or in progress.	SITE EMERGENCY
Total loss of Annuniciator System, computer alarms, and sequence of events recorder. (Not applicable when plant is at or below cold shutdown.)	Most or all alarms (annunciators) lost.	ALERT
Significant loss of ESF or Rx Protection instrumentation. An Unusual Event should NOT be declared for a non-emergency Tech Spec backdown,when the affected parameter remains monitorable. (Not applicable when plant is at or below cold shutdown.)	Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment capability.	UNUSUAL EVENT

TABLE 4-1 CHART H

(DELETED)

TABLE 4-1 CHART 1
SECONDARY SIDE ANOMALY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Main steam line break that results in a SI actuation **AND** (2) a. R-15 or R-19 reads offscale high with confirmation by chemistry analysis OR b. Primary to secondary leakage >50 gpm. **AND** (3) a. R-9 or CNTMT high range rad monitors (42599, 42600) indicate >10 R/hr OR b. CNTMT hydrogen monitor indicates >1% hydrogen concentration.	Steam line break **AND** primary to secondary leak >50 GPM **AND** Indication of Fuel Damage.	SITE EMERGENCY
Main steam line break that results in a SI actuation **AND** a. R-15 **OR** R-19 reads a factor of 1000 above normal **OR** b. Primary to secondary leakage >10 gpm.	Steam line break with significant (greater than 10 GPM) primary to secondary leakage. (Applies even if events occur in opposite steam generators.)	ALERT
Turbine trip and observation of penetration of casing.	Turbine rotating component failure causing rapid plant shutdown.	UNUSUAL EVENT
The uncontrolled depressurization of the secondary system to <500 psig steam generator pressure (SI actuation setpoint).	Rapid depressurization of the secondary side.	UNUSUAL EVENT

TABLE 4-1 CHART J
MISCELLANEOUS ABNORMAL PLANT CONDITIONS

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Containment boundary failure or potential failure a. Containment pressure >46 psig **OR** b. Loss of all containment fancoil units and both trains of ICS **OR** c. Containment hydrogen monitor ≥10% hydrogen concentration **AND** (2) Loss of core cooling capability a. Loss of SI and RHR flow **AND** (3) Failure of shutdown system when required: a. Entry into IPEOP FR-S.1, "Response to Nuclear Power Generation/ATWS" **OR** b. Loss of AFW for greater than 30 minutes with loss of main FW and condensate.	Other plant conditions that make a release of large amounts of radioactivity in a short time period possible; e.g., any core melt situation. Examples: - Failure of main FW and AFW systems for greater than 30 minutes without Safety Injection and Residual Heat Removal flow. Plus a containment failure is imminent. - Transient requiring the operation of shutdown systems with a failure of these shutdown systems. In addition, failure of SI and RHR and containment failure is imminent.	GENERAL EMERGENCY
Evacuation of Control Room (E-O-06 event).	Evacuation of control room and control of shutdown systems required from local stations.	SITE EMERGENCY
Conditions that warrant increased awareness on part of the plant staff will be evaluated by the Plant Manager or his designate. This is to determine if conditions are applicable for activating the E.P. <u>Example</u>: Loss of AFW system when required, validated upon implementation of FR H.1 "Response to Loss of Secondary Heat Sink".	Other plant conditions that warrant increased awareness on the part of plant staff or state and/or local anthorities.	UNUSUAL EVENT

TABLE 4-1 CHART K
FIRE AND FIRE PROTECTION

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
A fire within the Auxiliary Building, Technical Support Center, safeguards alley, D/G rooms or screenhouse that defeats redundant safety trains of ESF equipment causing the required ESF system to be inoperable.	A fire compromising the functions of safety systems.	SITE EMERGENCY
A fire within the Auxiliary Building, Technical Support Center, safeguards alley, D/G rooms or screenhouse that causes a single train of required ESF equipment to be inoperable.	A fire potentially affecting safety systems.	ALERT
A fire within the Administration Building, Technical Support Center, Turbine Building, Warehouse-Annex, Auxiliary Building, or Containment Building lasting more than 10 minutes.	A fire within the plant lasting more than 10 minutes.	UNUSUAL EVENT

TABLE 4-1 CHART L

(DELETED)

TABLE 4-1 CHART M
EARTHQUAKE

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Activation of seismic recorder with TRIGGER, OBE, and DBE lights lit in relay room on RR159 AND (2) Verification of a seismic event by physical experience or from U. of W. - Milwaukee Seismic Center.	An earthquake greater than Design Basis Earthquake (DBE).	SITE EMERGENCY
(1) Activation of seismic recorder with TRIGGER, and OBE lights lit in relay room on RR159 AND (2) Verification of a seismic event by physical experience or from U. of W. - Milwaukee Seismic Center.	An earthquake greater than Operational Basis Earthquake (OBE).	ALERT
a. Activation of seismic recorder with TRIGGER light lit in relay room on RR159 OR b. An earthquake felt in the Plant*. (*Should be confirmed by evidence of physical damage or verification from University of Wisconsin Seismic Center.)	An earthquake felt in plant or detected on station seismic instrumentation.	UNUSUAL EVENT

NOTE: Telephone numbers for U of W - Milwaukee Seismic Center are in EPIP APPX-A-3.

TABLE 4-1 CHART N
HIGH WINDS OR TORNADO

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
(1) Winds in excess of 100 mph for greater than 1 hour AND (2) Plant above cold shutdown condition.	Sustained winds in excess of design levels with plant not in cold shutdown.	SITE EMERGENCY
(1) A tornado which strikes the facility AND (2) Causes damage to render a single train of required ESF equipment to be inoperable.	Any tornado striking facility.	ALERT
A tornado observed on-site causing significant damage to the facility.	Any tornado on-site.	UNUSUAL EVENT

TABLE 4-1 CHART O
FLOOD, LOW WATER, OR SEICHE

KNPP INDICATION				EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
FOREBAY LEVEL				Flood, low water, or seiche near design levels.	ALERT
0 PUMPS	1 PUMP	2 PUMPS	CORRESPOND TO LAKE LEVEL		
NOTE 3	NOTE 1	≥94% *	≥588 ft.		
≤64% *	≤42% *	≤42% *	≤573 ft.		
OR Deep water Wave ≥22.5 ft.					
FOREBAY LEVEL				50-year flood, low water or seiche.	UNUSUAL EVENT
0 PUMPS	1 PUMP	2 PUMPS	CORRESPOND TO LAKE LEVEL		
NOTE 2	≥98% *	≥88% *	≥586 ft.		
≤71% *	≤63% * NOTE 4	≤54% * NOTE 4	≤575 ft. 4 in.		
OR Deep water wave ≥18 ft.					

NOTE 1: Above the bottom of bar No. 1 painted on the south wall of the forebay.

NOTE 2: Above the bottom of bar No. 2 painted on the south wall of the forebay.

NOTE 3: Above the bottom of bar No. 3 painted on the south wall of the forebay.

NOTE 4: Applies to an uncontrollable decrease (cannot be restored by operator action; e.g., throttling water box valves, etc.).

* Computer point for forebay level is L9075A and should be used because of its greater accuracy.

Plant elevations and lake elevations are referenced to International Great Lakes Datum (IGLD), 1955.

(IGLD 1955 = IGLD 1985 - .7 FEET)

TABLE 4-1 CHART P
EXTERNAL EVENTS AND CHEMICAL SPILLS

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
An aircraft crash into plant buildings which causes a complete loss of an ESF function.	Aircraft crash affecting vital structures by impact OR fire.	SITE EMERGENCY
A missile strikes plant buildings or explosion occurs within a plant building which causes a complete loss of an ESF function.	Severe damage to safe shutdown equipment from missiles or explosion.	SITE EMERGENCY
Release of flammable or toxic gas from a ruptured container which causes or is likely to cause evacuation of stations necessary to control shutdown systems. Portable monitors indicate explosive or toxic concentrations of the gas at life threatening levels in those vital areas.	Uncontrolled release of toxic or flammable gas is confirmed within vital area.	SITE EMERGENCY
An aircraft crashes into plant buildings and causes a single train of required ESF equipment to be inoperable.	Aircraft crash on facility.	ALERT
A missile strikes the facility and causes a single train of required ESF equipment to be inoperable.	Missile impact from whatever source on facility.	ALERT
Release of toxic or flammable gas at life threatening levels from a ruptured container enter the protected area AND impacts safe operation of the plant.	Uncontrolled release of toxic or flammable gas is confirmed within the protected area.	ALERT
Self-explanatory.	Known explosion damage to facility affecting plant operation.	ALERT
(1) An aircraft crash within the site boundary OR (2) Unusual aircraft activity such as erratic flying, dropped unidentified object, or other hostile acts which threaten the plant or plant personnel. (Any other persistent aircraft activity for which identification attempts through the FAA or other agencies have been unsuccessful.)	Aircraft crash on-site or unusual aircraft activity over facility.	UNUSUAL EVENT
Release of toxic or flammable gas from a ruptured tank/truck on site. Portable monitors indicate toxic or explosive concentrations at life threatening levels of the gas near the spill area.	Uncontrolled release of toxic or flammable gas is confirmed on site.	UNUSUAL EVENT

TABLE 4-1 CHART Q
SECURITY CONTINGENCY

KNPP INDICATION	EMERGENCY CLASSIFICATION CRITERIA	CLASSIFICATION
Physical attack on the plant that has resulted in unauthorized personnel occupying the control room or any other vital areas as described in the Security Plan.	Loss of physical control of the plant.	GENERAL EMERGENCY
Physical attack on the plant involving imminent occupancy of the control room, auxiliary shutdown panels, or other vital areas as defined by the Security Plan.	Imminent loss of physical control of the plant.	SITE EMERGENCY
Security safeguards contingency event that results in adversaries commandeering an area of the plant, but not control over shutdown capability or of any vital areas as defined in the Security Plan.	Ongoing security compromise.	ALERT
Examples:- Bomb threat accompanied by interception of bomb materials. -Adversary intercepted in the protected area. -Undetonated bomb found on premises.	Security threat or attempted entry or attempted sabotage.	UNUSUAL EVENT

NOTE: Security staff will **not** act as notifier during **security events**. Utilize Control Room staff for notifications.

TABLE 4-2

CLASSIFICATION OF POSTULATED ACCIDENTS

These events are based upon the worst case conditions described in Chapter 14 of the USAR for the Kewaunee Nuclear Power Plant. To fully understand the event, the USAR must be consulted.

EVENT	EMERGENCY CLASSIFICATION	USAR SECTION
(1) Uncontrolled RCCA withdrawal from a subcritical condition.	*	14.1.1
(2) Uncontrolled RCCA withdrawal at power.	*	14.1.2
(3) RCC assembly misalignment.	*	14.1.3
(4) Chemical and Volume Control System malfunction.	*	14.1.4
(5) Start-up of a inactive reactor coolant loop.	*	14.1.5
(6) Excessive heat removal due to Feedwater System malfunctions.	*	14.1.6
(7) Excessive load increase incident.	*	14.1.7
(8) Loss of reactor coolant flow lock rotor of RC pump.	ALERT	14.1.8
(9) Loss of external electrical load.	*	14.1.9
(10) Loss of normal feedwater.	*	14.1.10
(11) Anticipated transient without scram.	ALERT	14.1.11
(12) Loss of AC power to the plant auxiliaries.	UNUSUAL EVENT	14.1.12
(13) Fuel handling accidents major failure of one element's cladding.	SITE EMERGENCY	14.2.1
(14) Accidental release - recycle or waste liquid.	*	14.2.2
(15) Accidental release - water gas Gas decay tank rupture Volume control tank rupture	 UNUSUAL EVENT UNUSUAL EVENT	14.2.3

NOTE

* The immediate results of these events taken alone are less than the criteria for notification of an unusual event.

SECTION 5

5.0 ORGANIZATIONAL CONTROL OF EMERGENCIES

Using the WPSC nuclear organization as a base, this section of the plan describes the overall emergency organization that would be used during emergency situations at the plant. This section delineates the responsibilities and assignments of plant and corporate emergency response personnel and describes their functional areas of emergency response activities. The latter part of this section describes the emergency response functions of Federal, state, local and private organizations.

5.1 NORMAL NUCLEAR ORGANIZATION

The Senior Vice President - Nuclear Power located at the WPSC corporate office has overall responsibility for the WPSC Nuclear Power and Quality Programs organizations. Reporting to the Senior Vice President - Nuclear Power are the Manager - Nuclear Plant Support Services, Manager - Kewaunee Plant, Manager - Engineering and Technical Support, Manager - Nuclear Business Group, and Superintendent - Quality Programs, and Nuclear Communications Coordinator (see FIGURE 5-1.1).

The Manager - Nuclear Plant Support Services is located at the plant and is responsible for nuclear organization training, plant protective services, emergency preparedness, administrative support, and Human Resources (see FIGURE 5-1.2). This position provides management oversight of nuclear power production activities that support the implementation of the emergency preparedness program.

The Manager - Kewaunee Plant is located at the plant and is responsible for the day-to-day operation of the plant. This includes operations, maintenance, instrument and control, radiation protection, and radiochemistry (see FIGURE 5-1.3).

The Manager - Engineering and Technical Support is located at the plant and is responsible for day to day engineering support, evaluation of plant activities, engineering programs and projects, plant modifications, and day-to-day interaction with the NRC (see Figure 5-1.1).

The Manager - Nuclear Business Group is located at the WPSC corporate office and is responsible for legal and regulatory interaction, budgeting, purchasing, strategic planning, license renewal, high level waste, and decommissioning (see Figure 5-1.1).

The Superintendent - Quality Programs is located at the plant and is responsible for the Administration and Implementation of Quality Control Engineering Activities, Quality Control Activities, and ensuring the effective implementation of the WPSC Operational Quality Assurance Program (see FIGURE 5-1.1).

The Nuclear Communications Coordinator is located at the WPSC corporate office and is responsible for the Nuclear Emergency Public Information Plan, the daily public information, media relations, nuclear employee communications, and external education and information services of the Nuclear Department (see Figure 5-1.1).

The Kewaunee Plant organization is on-site during regular working hours, Monday through Friday, holidays excluded, with the following exceptions; the plant operating shift organization, which includes operations, radiation protection, and security personnel, are on duty on a 24-hour basis, the chemistry needs of the plant are normally fulfilled by using an 18-hour shift schedule, but as a minimum, provides day and evening shifts on Monday through Friday, and day shift only on weekends and holidays. The following subsection describes this plant operating shift organization.

5.1.1 Plant Operating Shift Organization

The plant operating shift staff consists of eight plant staff personnel and an appropriate number of security personnel. The Shift Supervisor, who holds a Senior Reactor Operator (SRO) license, is in direct charge of all plant operations during his assigned shift and is responsible for the supervision and actions of the operating personnel on the shift. The Shift Supervisor will be assisted by a Control Room Supervisor who also holds an SRO license. Additional shift personnel include: two Nuclear Control Operators who hold Reactor Operator (RO) licenses, two Nuclear Auxiliary Operators, a Radiation Technologist, a Shift Technical Advisor, and a Chemistry Technologist (per the shift schedule stated in 5.1 above). The duties and responsibilities of the operating staff are defined in the Nuclear Administrative Directives. In addition, a Security Shift Captain, with supporting security officers, fill security positions as well as fill the position of Notifier during declared emergencies until a designated Control Room Communicator reports to the Control Room. FIGURE 5-2 shows the composition of the plant operating shift organization.

5.2 EMERGENCY RESPONSE ORGANIZATION

In the event of a declared emergency, appropriate groups of the emergency response organization shall be activated. The pre-assignment of plant and corporate personnel to key functional areas of emergency activities ensures automatic, unambiguous manning and coordination of the emergency response organization and immediate response capabilities during emergency situations.

The emergency response organization can be activated during normal or off-normal working hours. During normal working hours, the emergency response organization will be formed through transition of the normal WPSC nuclear organization (see FIGURES 5-1 and 5-2) into an emergency mode of operation depending on the situation and emergency classification. During off-normal working hours, the emergency response organization shall consist of the plant operating shift staff (see FIGURE 5-2) augmented by additional members of the plant and corporate nuclear staff as required.

To augment the plant operating shift staff with additional personnel in an emergency, plant and corporate emergency response personnel are provided with radio pagers. It has been established that emergency response personnel not on site at the initiation of an emergency could begin to arrive approximately 15 minutes after notification that an emergency has been declared at the Kewaunee Nuclear Power Plant. Emergency response personnel are pre-assigned and cross-trained to meet the functional staffing requirements stipulated in Table B-1 of NUREG-0654/FEMA-REP-1, Revision 1.

The following subsections describe the pre-assigned emergency responsibilities of WPSC plant and corporate headquarters personnel for events classified as an Unusual Event, Alert, Site Emergency, or General Emergency. FIGURE 5-3 shows the overall WPSC emergency organizational structure for the Kewaunee Nuclear Power Plant. Emergency Plan Appendix A provides the emergency titles, the locations, and the primary responsibilities of the key emergency response personnel. Emergency Plan Implementing Procedure, Appendix A-2, "Response Personnel Call List" correlates emergency organization job titles with the qualified individual who can fill those positions.

FIGURE 5-1.1

WPSC NUCLEAR ORGANIZATION

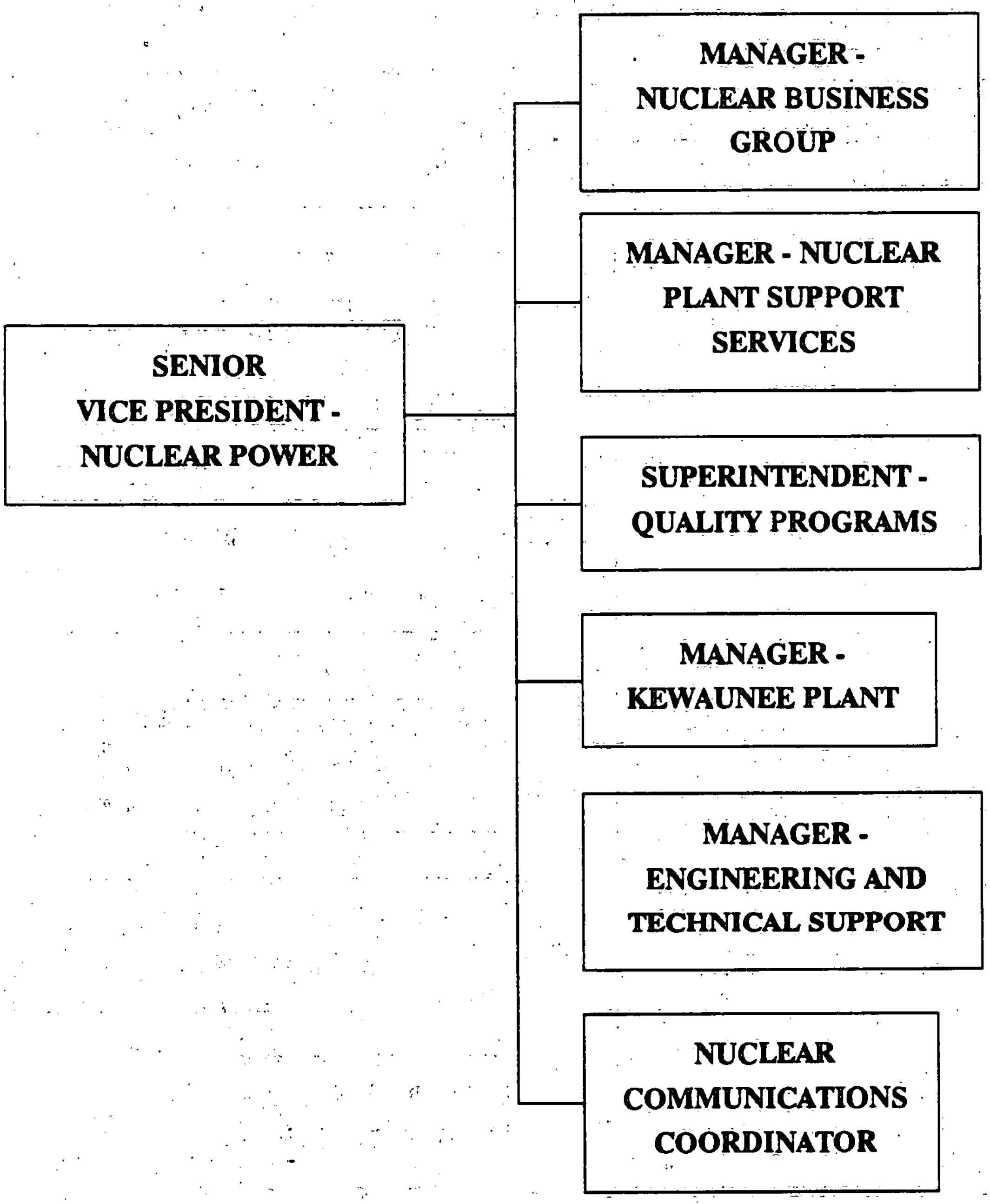


FIGURE 5-1.2

WPSC NUCLEAR ORGANIZATION

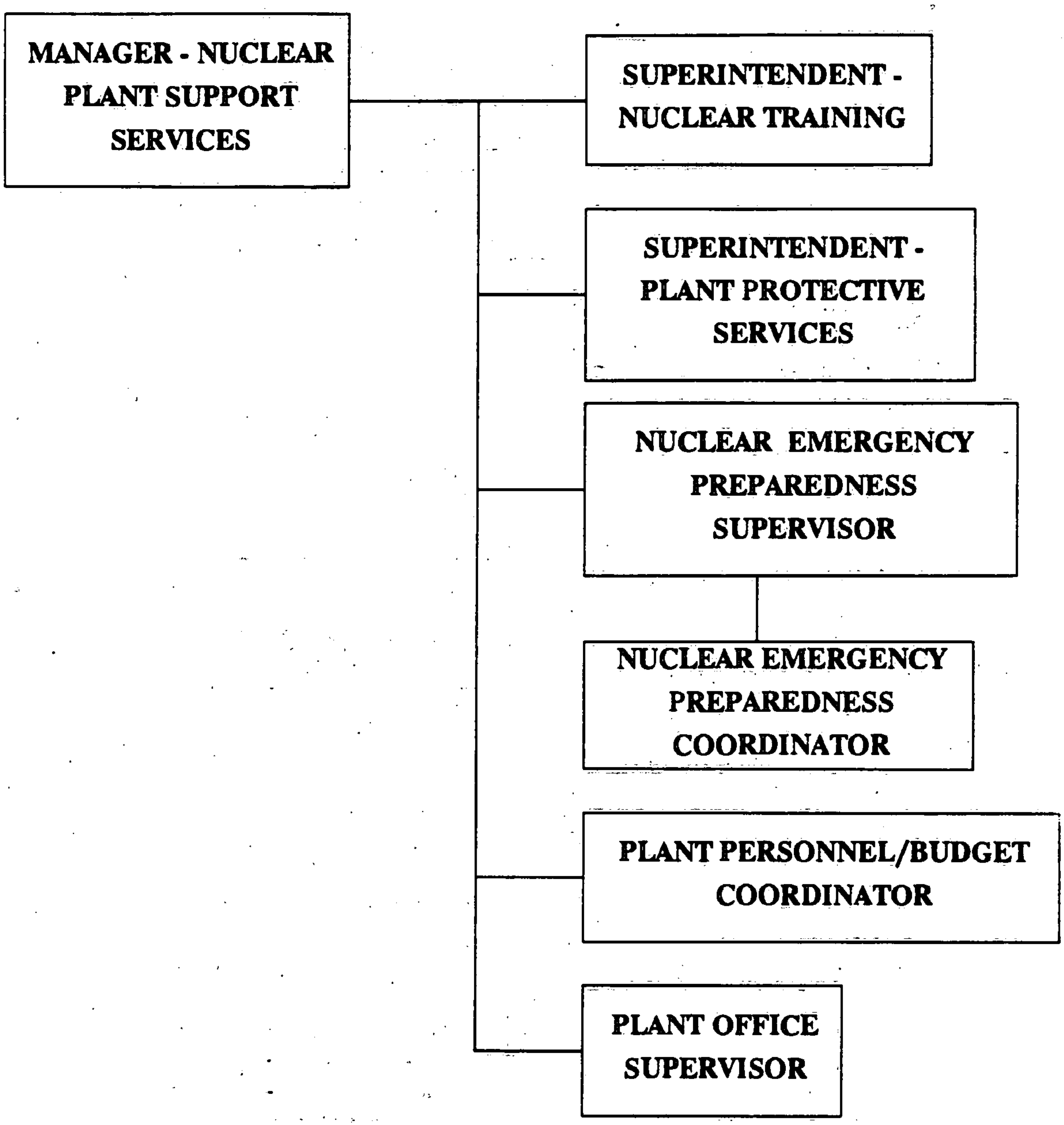


FIGURE 5-1.3

WPSC NUCLEAR ORGANIZATION

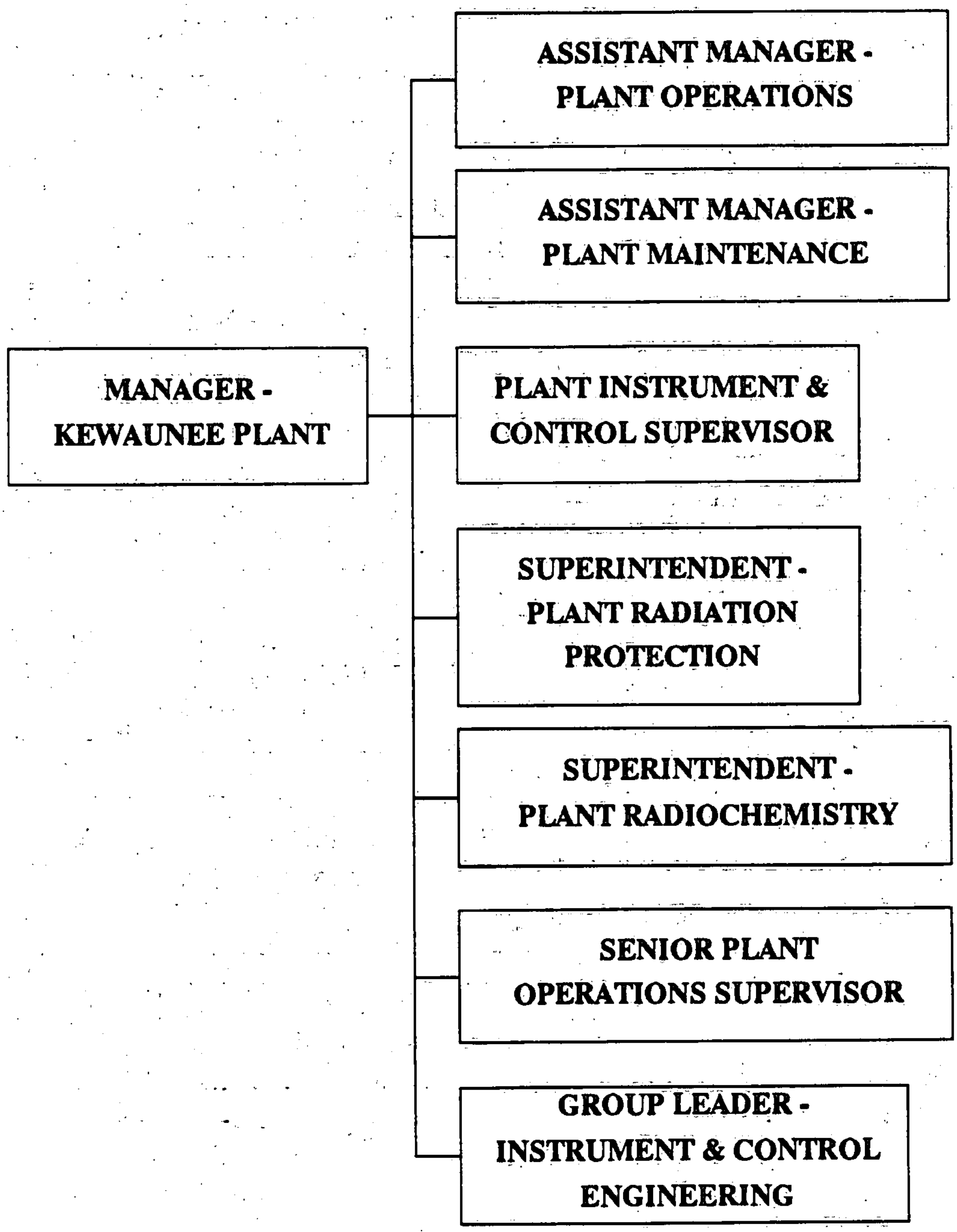


FIGURE 5-1.4

WPSC NUCLEAR ORGANIZATION

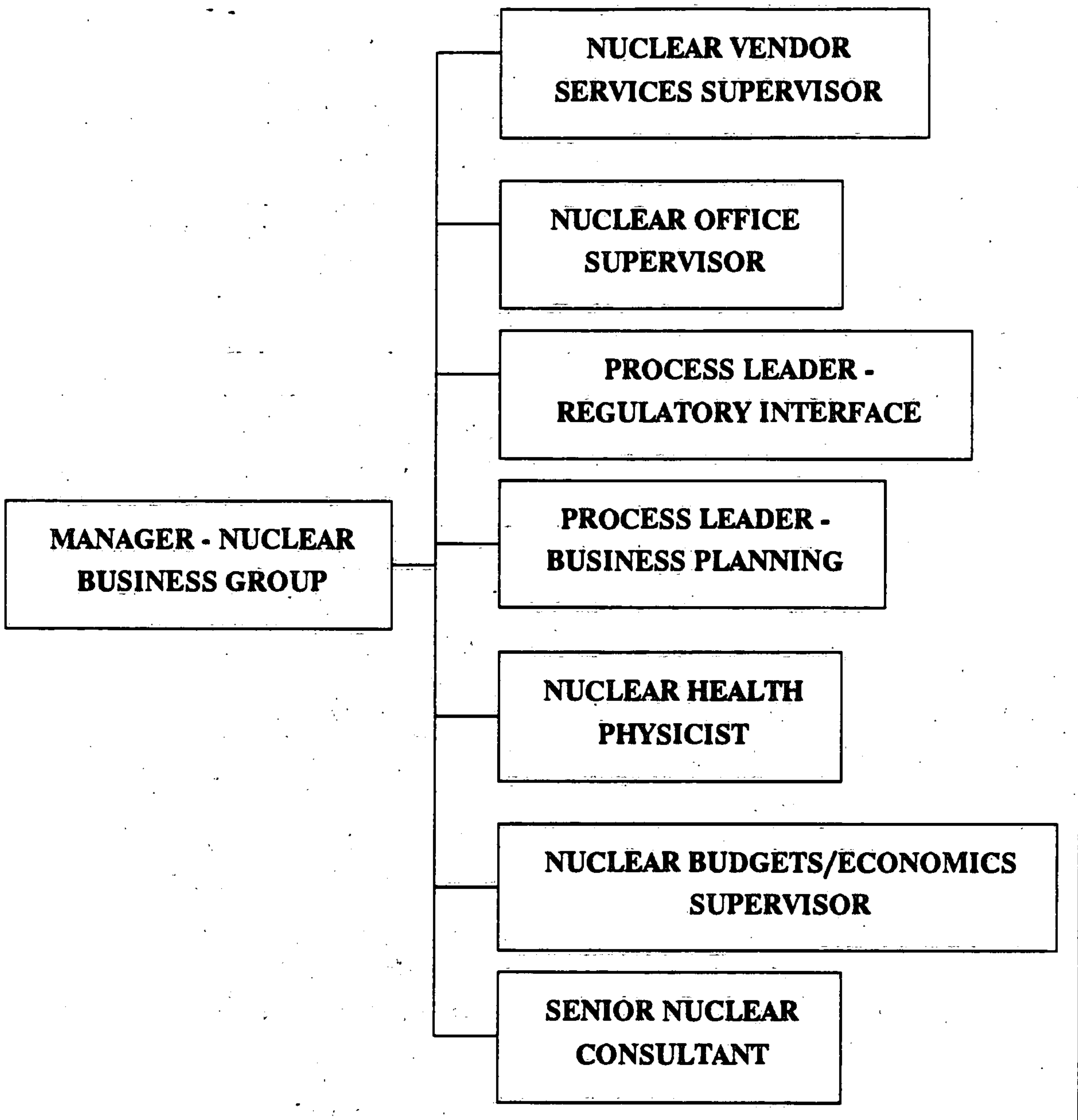


FIGURE 5-1.5

WPSC NUCLEAR ORGANIZATION

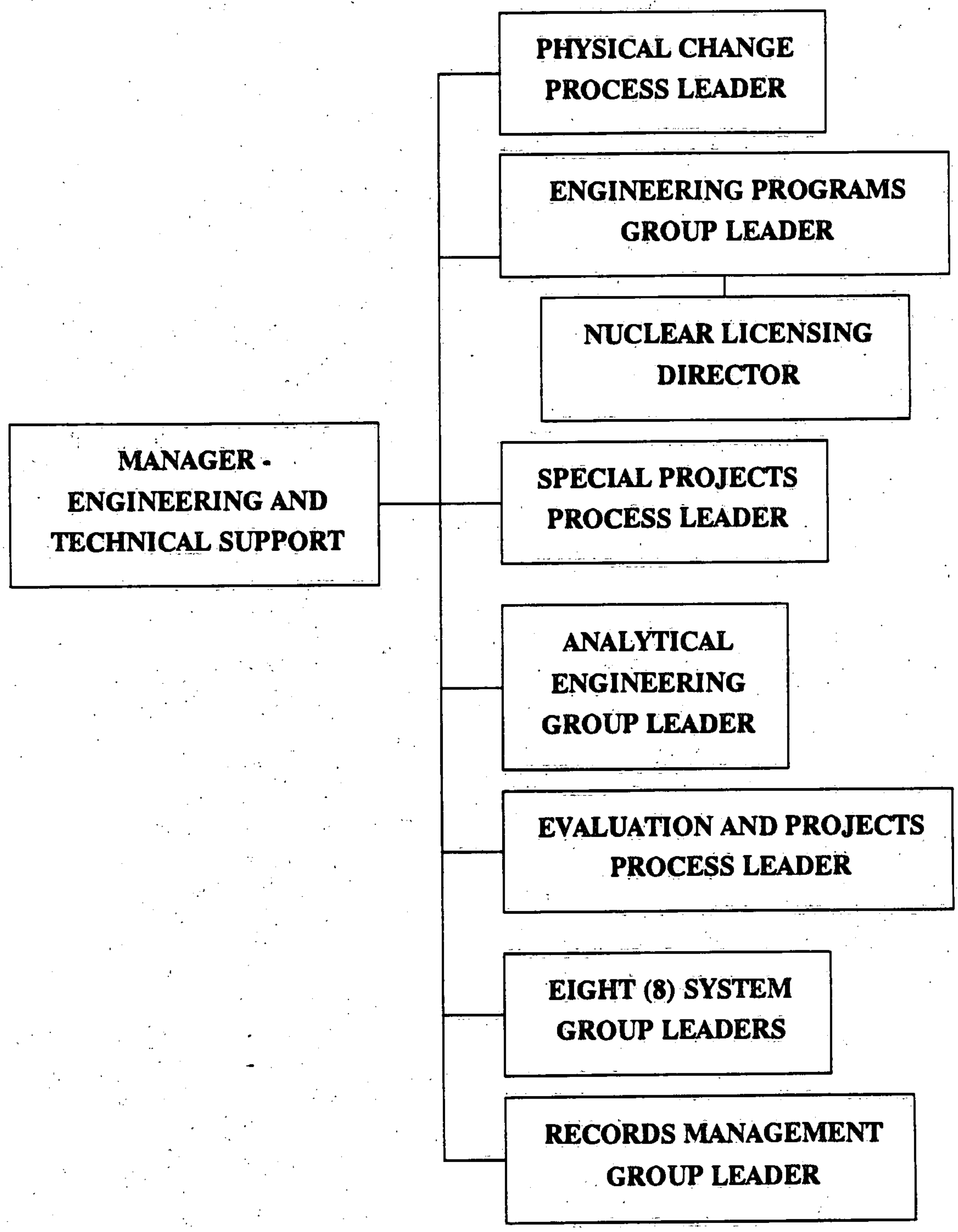
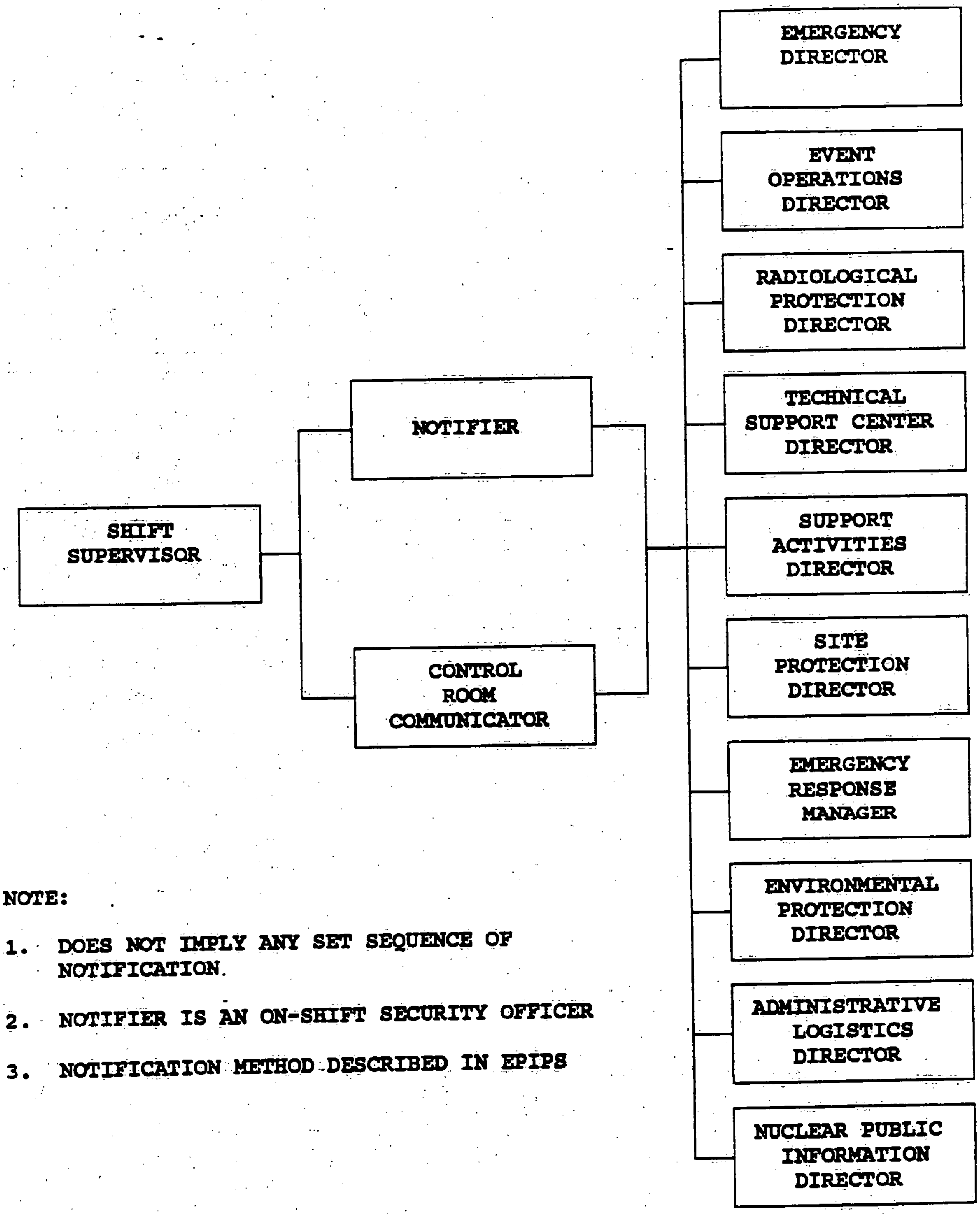


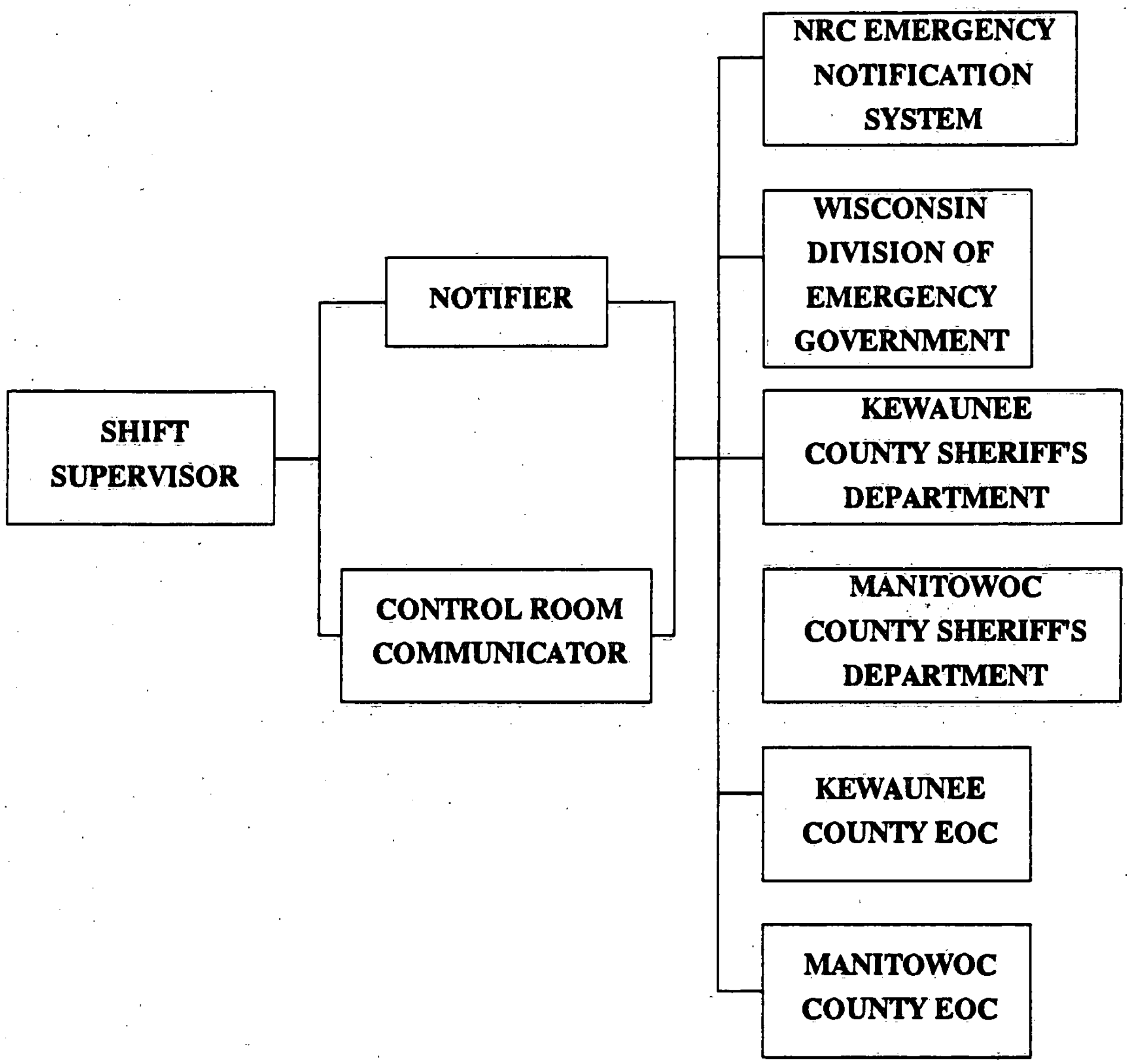
FIGURE 6-1
PLANT AND CORPORATE NOTIFICATION



NOTE:

1. DOES NOT IMPLY ANY SET SEQUENCE OF NOTIFICATION.
2. NOTIFIER IS AN ON-SHIFT SECURITY OFFICER
3. NOTIFICATION METHOD DESCRIBED IN EPIPS

FIGURE 6-2
INITIAL OFF-SITE NOTIFICATION



NOTE:

1. DOES NOT IMPLY ANY SET SEQUENCE OF NOTIFICATION
2. NOTIFIER IS AN ON-SHIFT SECURITY OFFICER
3. NOTIFICATION METHOD DESCRIBED IN EPIPS

7.1 EMERGENCY RESPONSE FACILITIES

Several emergency response facilities have been established to support emergency response operations (see Appendix C, FIGURE C-6 and C-7 for locations). These emergency response facilities operate as a coordinated group but are physically separated to minimize interference and confusion. Dedicated communication lines between the facilities ensure an uninterrupted flow of data and instructions. The emergency response facilities contain water, sanitary and other provisions for use by emergency personnel. Supplementary services, such as food and additional equipment, are readily obtainable thus ensuring the capability of long term, uninterrupted emergency response operations. Appendix F lists the emergency equipment and materials located in each emergency response facility. A detailed list of Control Room equipment and instrumentation is provided in Section 7 of the Updated Safety Analysis Report (USAR).

7.1.1 Control Room

The Control Room is the primary facility at the Kewaunee Nuclear Power Plant in which plant conditions are monitored and controlled and corrective actions are taken to mitigate any abnormal occurrence. It is operated under the direction of the Shift Supervisor and is the location where initial assessment, emergency classification and emergency response begins.

The controls and instrumentation necessary to operate the plant under both normal and emergency conditions are located in the Control Room. The Control Room is equipped with plant parameter instrumentation such as area and process radiation monitoring systems and alarm annunciators that give early warning of a potential emergency and provide for a continuing evaluation of the emergency situation. Additional equipment such as portable radiation survey instruments, meteorological readouts and communications equipment are also located in the Control Room. The Control Room has communications capability with all on-site and off-site emergency response facilities via the plant PBX phone system. Should the PBX system fail, other non-PBX system phone lines are installed in the Control Room (dedicated incoming lines). In addition, the Control Room has communications capability with all on-site emergency response facilities via the in-plant public address system, and communications capability with off-site state and local authorities via Dial Select System. The Control Room also has the capability to communicate with the NRC over the Emergency Notification System.

The Control Room is designed to be habitable under emergency conditions. The ventilation system, shielding and structural integrity of the Control Room permit continuous occupancy during postulated design basis accidents described in Section 14 of the USAR.

7.1.2 Technical Support Center

The Technical Support Center (TSC) is located north of and adjacent to the Auxiliary and Turbine Buildings. This location is in close proximity to the Control Room. It is approximately 4,000 ft^2 in area and capable of accommodating more than 25 people. Plant engineering data and safety parameter displays to support Control Room operations are installed in the TSC.

The TSC is activated upon the declaration of an Alert, Site Emergency or General Emergency. It operates under the direction of the TSC Director and serves as the coordination point for technical support during emergency response operations. The TSC provides the communications interface between the Control Room, the Radiological Analysis Facility, the Operational Support Facility, the Emergency Operations Facility and the Site Boundary Facility. Follow-up communications with Federal, state and local response organizations will be coordinated in the TSC prior to the activation of the Emergency Operations Facility (EOF). The TSC has communications capabilities with all on-site and off-site emergency response facilities via the plant PBX phone system. Should the PBX system fail, additional non-PBX phone lines are installed in the TSC (dedicated incoming lines). In addition, the TSC also has communications capability with all on-site emergency response facilities via the in-plant public address system. The TSC also has direct (dial-select) communication lines to the Emergency Operations Centers (EOC) for both Kewaunee and Manitowoc counties; the State of Wisconsin EOC in Madison; the Point Beach Nuclear Plant's TSC and EOF; and the Kewaunee Nuclear Power Plant's EOF. The TSC also has the capability to communicate with the NRC over the Emergency Notification System and Health Physics Network telephones.

Adequate equipment exists in the TSC to provide the TSC staff with the capability to monitor reactor systems status and to evaluate plant system abnormalities. This equipment includes signal display instrumentation, data displays and information storage and retrieval devices. The data displays will provide current indications and time history displays of plant parameters. A remote terminal will have the capability of displaying selected data from the plant process computer.

The TSC staff will provide information on radiological process and effluent monitors to the Radiological Analysis Facility for use in predicting radiological consequences in addition to analyzing plant data and information to make recommendations to the Emergency Director concerning accident mitigation and recovery operations.

The TSC is designed to have the same radiological habitability as the Control Room under accident conditions and has permanent monitoring systems which indicate radiation dose rates and airborne radioactivity concentrations. The air purification system design includes particulate and charcoal filters to meet post-accident habitability requirements.

7.1.3 Radiological Analysis Facility and Radiation Protection Office

The Radiological Analysis Facility (RAF) operates in conjunction with the Radiation Protection Office (RPO) in coordinating and directing radiation protection activities. The RAF and RPO are activated during an Alert, Site Emergency, or General Emergency. These facilities are operated under the direction of the Radiological Protection Director.

The RPO is located in the Auxiliary Building and serves as the normal access control point for the Radiological Controlled Area and as an assembly area for personnel accountability purposes. It is the headquarters for plant radiation protection activities which include radiological surveys, personnel monitoring, decontamination, reentry, and rescue operations. The RPO is equipped with radiation monitoring and sampling equipment, protective clothing, respiratory protection devices, and other miscellaneous supplies for use during emergency situations. The RPO has the capability to communicate with all on-site and off-site emergency response facilities via the plant PBX phone system and with all on-site emergency response facilities via the in-plant public address system.

The RAF is located adjacent to the TSC. The RAF is the central location for directing plant radiological activities during emergency situations. Survey equipment, maps, and radiocounting equipment are available in the RAF for making dose projections and for tracking gaseous and liquid effluents.

The RAF serves as an emergency access point into the Auxiliary Building. Since the RAF is in the same building as the TSC, it will be habitable throughout the duration of an incident. The RAF has communications capability with all on-site and off-site emergency response facilities via the plant PBX phone system, and with all on-site emergency response facilities via the in-plant public address system. In addition, the RAF has the capability to communicate with the NRC over the Health Physics Network System.

7.1.4 Operational Support Facility

The Operational Support Facility (OSF) is located adjacent to the TSC. The OSF is activated during an Alert, Site Emergency or General Emergency. It is operated under the direction of the Support Activities Director and is where operational and maintenance support personnel report for emergency assignment or assembly when they are not actively engaged in emergency duties. The OSF serves as a staging area for briefing plant maintenance and non-shift operating personnel. The OSF has communications capability with all on-site and off-site emergency response facilities via the plant PBX phone system. In addition, the OSF has communications capability with all on-site emergency response facilities via the in-plant public address system. Since the OSF is in the same building as the TSC, it will be habitable throughout the duration of an incident.

7.1.5 Emergency Operations Facility

The Emergency Operations Facility (EOF) is located in the WPSC Green Bay Division Building, in the city of Green Bay, Wisconsin.

The EOF is activated during an Alert, a Site Emergency or a General Emergency. It is operated under the direction of the Emergency Response Manager. The EOF has adequate space to accommodate representatives from various Federal, State and local organizations.

The EOF is the focal point for the coordination of on-site and off-site emergency response activities. Management and technical personnel assigned to the EOF are responsible for protective action recommendations, liaison with off-site governmental organizations and response facilities and overall management of the emergency organization.

The EOF can communicate with all on-site and off-site emergency response facilities via the WPSC Corporate Office PBX system. Should the PBX system fail, additional non-PBX phone lines are installed in the EOF. In addition, the EOF has direct (dial-select) communication links to; the Kewaunee and the

Manitowoc County Emergency Operations Centers (EOC); the Point Beach Nuclear Plant's TSC and EOF; the State of Wisconsin's EOC in Madison; and Kewaunee Nuclear Power Plant TSC. Branches of the Emergency Notification System and Health Physics Network System are available in the EOF.

7.1.6 Joint Public Information Center

The Joint Public Information Center (JPIC) is located adjacent to the WPSC Green Bay Division Building in the city of Green Bay, Wisconsin. The JPIC is activated during an Alert, a Site Emergency, a General Emergency or at the direction of the Emergency Response Manager. The Nuclear Public Information Director supervises WPSC activities at the JPIC and assists the WPSC Corporate Spokesperson. The JPIC is utilized to formulate and coordinate the development of news statements for the news media concerning the emergency. This facility provides periodic updates of the emergency situation and coordinates the general public information activities of WPSC and the appropriate Federal, state and local agencies to ensure that only authorized news statements are released. The general public shall be provided with a telephone number to call for the latest information regarding plant conditions. A WPSC corporate spokesperson shall be located at the JPIC to receive information from the EOF concerning plant status. The Spokesperson shall coordinate the information with the Nuclear Public Information Director.

7.1.7 Site Boundary Facility

Upon the declaration of an Alert, Site Emergency, or General Emergency, the Site Boundary Facility (SBF), located near the site boundary, west of State Highway 42, shall be activated to serve as a staging area for off-site environmental monitoring. Environmental monitoring and sample results shall be relayed to the Environmental Protection Director at the EOF. As radiological conditions require, and at the direction of the Emergency Director, the SBF may be used as the coordinating center for access control if the Security Building is not available. Radiological monitoring of personnel and equipment entering and leaving the site can be performed at the SBF. It will be staffed with environmental monitoring team, site radiation emergency team or security force members appropriate with emergency conditions.

The SBF has telephone communications via the plant PBX system, and radio communications to the RAF and EOF. It is equipped with emergency radiation monitoring, counting and sampling equipment, protective clothing, and other supplies for use during an emergency.

7.2 COMMUNICATION SYSTEMS

A comprehensive communication system with back-up capabilities has been designed to provide reliable communication links between various emergency response facilities and with off-site support organizations. The system consists of the plant PBX telephone system, the plant public address system, commercial telephone lines, a two-digit ring-down telephone network, a radio pager system, radio communications, and the National Warning System (NAWAS). The details of the site and off-site emergency communication networks are illustrated in FIGURES 7-1 and 7-2. A brief description of the communication systems is summarized below:

1. The stored program PBX telephone system at the Kewaunee Nuclear Power Plant is the primary and most reliable communications system used to transmit information and data between all the emergency response facilities. The overall reliability of the PBX system is established due to the following system design characteristics.

   1) The system is powered from an uninterruptable power supply.

   2) The system has an internal battery pack to supply power if the primary source is lost.

   3) The systems computer will automatically re-load the base program if the computer memory is for some reason lost. If there is a complete and total loss of the system there will still be at least seven (7) independent trunk lines available from an outside source.

2. The Plant public address system operates independent of the telephone system. The system has five paging channels and includes handset stations and loud speakers. The public address system has options for making general announcements or holding conversations via any of the five channels. Diesel generators serve as an emergency power source for the public address system.

3. Plant and corporate emergency response personnel have been issued pocket radio pagers. The radio paging transmitters at the plant site and corporate headquarters may be accessed via the plant or corporate PBX telephone system. Digital codes are used to access either individual or groups of pagers.

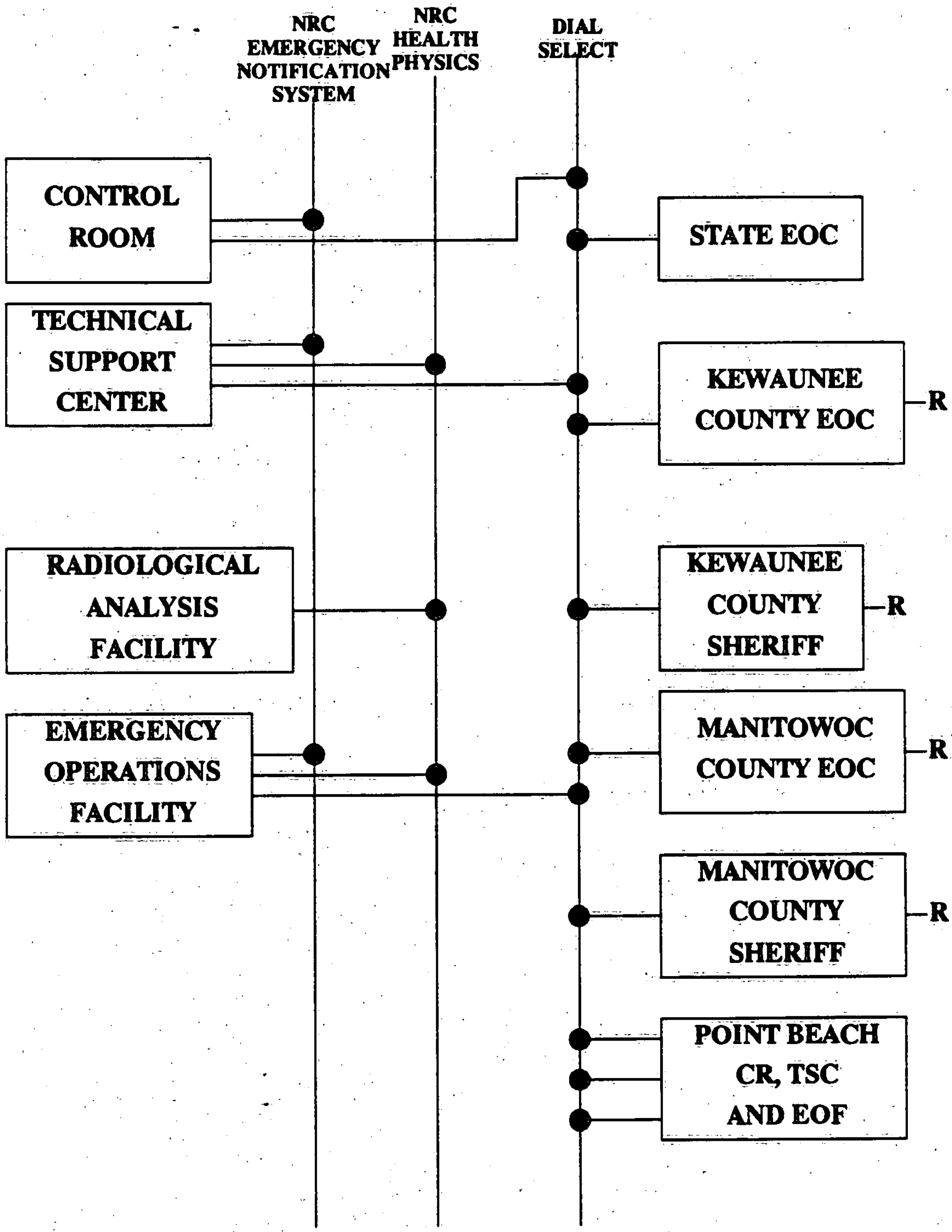
4. The Dial-Select Telephone System is the primary means for providing initial subsiquent notification of Declared Emergencies. This system provides a communication link with the Point Beach Nuclear Plant's Control Room, TSC, EOF and Alt-EOF; the Kewaunee and Manitowoc Counties' EOC's and Sheriffs Dispatch Centers; the State of Wisconsin EOC in Madison, the State Highway Patrol Dispatch Center and the Kewaunee Nuclear Power Plant's Control Room, TSC and EOF.

5. Direct phone lines have been installed to provide rapid, uninterrupted communication with the NRC. The Radiological Analysis Facility, the Technical Support Center and the Emergency Operations Facility have direct lines to the NRC Health Physics Network. The Control Room, the Technical Support Center, and the Emergency Operations Facility have dedicated lines into the NRC Emergency Notification System.

6. A radio base station is located in the Control Room, with remote console stations in the RAF and the EOF. The base and remote stations will be used to communicate with the Radiation Emergency Teams and Environmental Monitoring Teams. A transmit/receive capability exists, on an independent frequency, 24 hours a day from the Control Room to the Kewaunee County Sheriff's Department.

7. Dedicated commercial telephone lines are established to facilitate state and local authorities in contacting WPSC representatives. The first line allows direct access to the Control Room, Technical Support Center or Emergency Operations Facility, as appropriate, for state and local emergency government data verification calls. A second line is available to receive two calls at one time (hunting feature). The third line allows direct access to the Radiological Analysis Facility or Emergency Operations Facility, as appropriate, for the State Radiological Coordinator to obtain plant, meteorological and radiological information.

FIGURE 7-2
OFF-SITE COMMUNICATIONS NETWORK



- R = Radio Communication

SECTION 8

8.0 MAINTAINING EMERGENCY PREPAREDNESS

It is important that a state of emergency preparedness be maintained at all times at the Kewaunee Nuclear Power Plant. To ensure a state of readiness, the Kewaunee Nuclear Power Plant emergency preparedness program was designed to provide each of the following objectives:

1. Formal designation of management personnel responsible for the emergency preparedness program.

2. Establishment of an Emergency Preparedness Training Program.

3. Planning and conducting periodic drills and exercises.

4. Annual review and updating of the Emergency Plan and the Emergency Plan Implementing Procedures (EPIPs).

5. Routine calibration, maintenance and inventory of emergency equipment and supplies.

6. Establishment of a Public Information Program.

This section of the Emergency Plan summarizes the emergency preparedness program that has been established for the Kewaunee Nuclear Power Plant.

8.1 EMERGENCY PREPAREDNESS RESPONSIBILITIES

<u>The Senior Vice President - Nuclear Power</u> is responsible for emergency response planning. This responsibility includes ensuring that the overall emergency preparedness program is maintained and implemented as described in this plan.

<u>The Nuclear Emergency Preparedness Supervisor</u> reports to the Manager - Nuclear Plant Support Services which provides a line of communication to the Senior Vice President - Nuclear Power. The Nuclear Emergency Preparedness Supervisor is responsible for the following tasks:

1. Coordinate the development and implementation of the emergency preparedness program.

2. Ensure that all drill and exercise commitments stated in the plan are met.

3. Schedule, coordinate and monitor emergency preparedness training programs, drills and exercises for both WPSC response personnel and off-site supporting agencies.

4. Coordinate and monitor material readiness of all emergency response facilities, and procedures to ensure adequate preparedness in accordance with this plan.

5. Coordinate adequate personnel coverage for specific emergency duties to assure optimal manpower coverage.

6. Obtain and maintain agreements of understanding between WPSC and Federal, state, local, and private organizations so that an adequate level of emergency backup support is available.

7. Assure that the Emergency Plan and its Emergency Plan Implementing Procedures are reviewed and updated annually.

8. Provide a summary of all Emergency Plan revisions and a safety evaluation following the annual review.

9. Monitor the development of all EPIPs and ensure all EPIPs receive an adequate technical review. Review all EPIP revisions to prevent the compromise of other EPIPs or the Emergency Plan.

10. Assure that the WPSC Nuclear Emergency Public Information Plan is maintained and that the annual mailing of the public information brochure is accomplished.

11. Assure that post-drill and exercise critiques are performed in accordance with established procedures.

12. Develop and distribute action plans to address drill and exercise deficiencies.

13. Compile off-site agency/utility interface problems from QA audits and technical reviews and forwards them to State and county emergency government officials.

<u>The Nuclear Licensing Director</u> reports to the Senior Vice President - Nuclear Power through the Manager - Engineering and Technical Support and Engineering Programs Group Leader and is responsible for the following tasks:

1. Maintaining current knowledge of changes in Federal regulations, state and local emergency plans, and other guidance that impact emergency planning activities.

2. Reviews proposed changes to the Emergency Plan in accordance with the provisions of 10 CFR 50.54 (q).

3. Submit Emergency Plan and EPIP revisions to the NRC.

<u>The Superintendent - Nuclear Training</u> reports to the Senior Vice President - Nuclear Power through the Manager - Nuclear Plant Support Services and is responsible for developing and implementing a training program for all plant and corporate personnel having emergency responsibilities. Training topics include the Emergency Plan, the EPIPs, and the emergency support provided by Federal, state, local and off-site organizations. The Superintendent - Nuclear Training is responsible for the following tasks:

1. Provide training support for both on-site and off-site emergency response organization members.

2. Review and update the Emergency Preparedness Training Program to incorporate program changes based on deficiencies noted during drills and exercises as well as Emergency Plan and EPIP revisions.

The Manager - Nuclear Plant Support Services reports to the Senior Vice President - Nuclear Power. This manager in conjunction with the Manager - Kewaunee Plant, Manager - Engineering and Technical Support, and Nuclear Communications Coordinator is responsible for overseeing plant and corporate support effort for the emergency preparedness program. This includes support for training, scenario development, controller participation, and procedure and plan development and implementation.

The WPSC Nuclear Department Heads reporting to the Manager - Nuclear Plant Support Services, Manager - Kewaunee Plant, Manager - Engineering and Technical Support and Manager - Nuclear Engineering (see FIGURE 5-1.2 through 5-1.5) are responsible for the following tasks:

1. Develop and update the EPIPs and EPMPs assigned to them.
2. Assist the Superintendent-Nuclear Training to coordinate and provide emergency preparedness training on EPIPs.
3. Assist the Nuclear Emergency Preparedness Supervisor with the following aspects of emergency preparedness.
    a. Assignment of personnel to emergency response organization positions.
    b. Response facility floor plans, status boards and logistics.
    c. Response facility material and equipment inventories.

The Plant Operations Review Committee (PORC) is responsible for reviewing proposed revisions to the designated EPIPs and EPMPs.

<u>The WPSC Quality Programs Group</u> shall conduct an indpendent performance based audit of the Emergency Preparedness Program which includes the Emergency Plan. This audit shall be conducted in accordance with the Operational Quality Assurance Program, at least every 12 months in accordance with 10 CFR 50.54(t); and shall include an assessment of the adequacy of interface with state and local governments, drills, exercises, personnel capabilities, and procedures.

The lead auditor conducting this audit shall have no direct responsibility for the implementation of the emergency preparedness program. However, managers and process leaders responsible for or knowledgable of specific aspects of the Emergency Preparedness Program any request a technical review of specific areas of the program.

The results of the audit shall be formally documented in an audit report and retained for the life of the plant. This audit report shall be distributed to the appropriate WPSC management in accordance with Quality Assurance Directive 14.1, "Quality Assurance Audits."

State and local governments shall be notified of inadequacies involving WPSC/government interface. The Quality Programs Group shall forward WPSC/government interface issues to the Nuclear Emergency Preparedness Supervisor.

8.2 EMERGENCY RESPONSE PERSONNEL TRAINING

All aspects of emergency preparedness training administration are specified in the emergency preparedness training program maintained by the Kewaunee Nuclear Power plant Training group. This program identifies the level and the depth to which individuals are to be trained. In addition to this program, appropriate personnel will be trained in the areas of radiation protection, respiratory protection, and multi-media first-aid or its equivalent as part of the Kewaunee Nuclear Power Plant's general employee training program.

Plant and corporate emergency preparedness group staff should receive training on emergency preparedness topics annually if appropriate programs are available in the industry.

8.2.1 Emergency Preparedness Training Program

The Emergency Preparedness Training Program consists of five general categories. Together, these categories cover the training needs of not only WPSC employees who have assigned emergency response duties but also the training needs of visitors, vendors, and off-site agencies who may be on site at the time of an incident or who may respond to the Kewaunee Nuclear Power Plant in support of the Emergency Plan. The training requirements of individuals assigned emergency response duties are defined in this program. Individuals shall meet all the applicable training requirements prior to being assigned to an emergency position.

The five categories are as follows:

(1) **GET (General Employee Training)** shall be given to all personnel who are badged for unescorted access to the Kewaunee Nuclear Power Plant. Information pertaining to their safety and the safety of visitors under escort during the initial stages of a declared emergency shall be provided.

FIGURE C-4
POPULATION DISTRIBUTION
BY EVACUATION AREAS
KEWAUNEE NUCLEAR POWER PLANT
KEWAUNEE AND MANITOWOC COUNTIES

AREA/POPULATION

2	=	203
5N	=	700
5S	=	585
10N	=	3559
10NW	=	530
10W	=	803
10SW	=	1456
10SSW	=	2029
10S	=	663

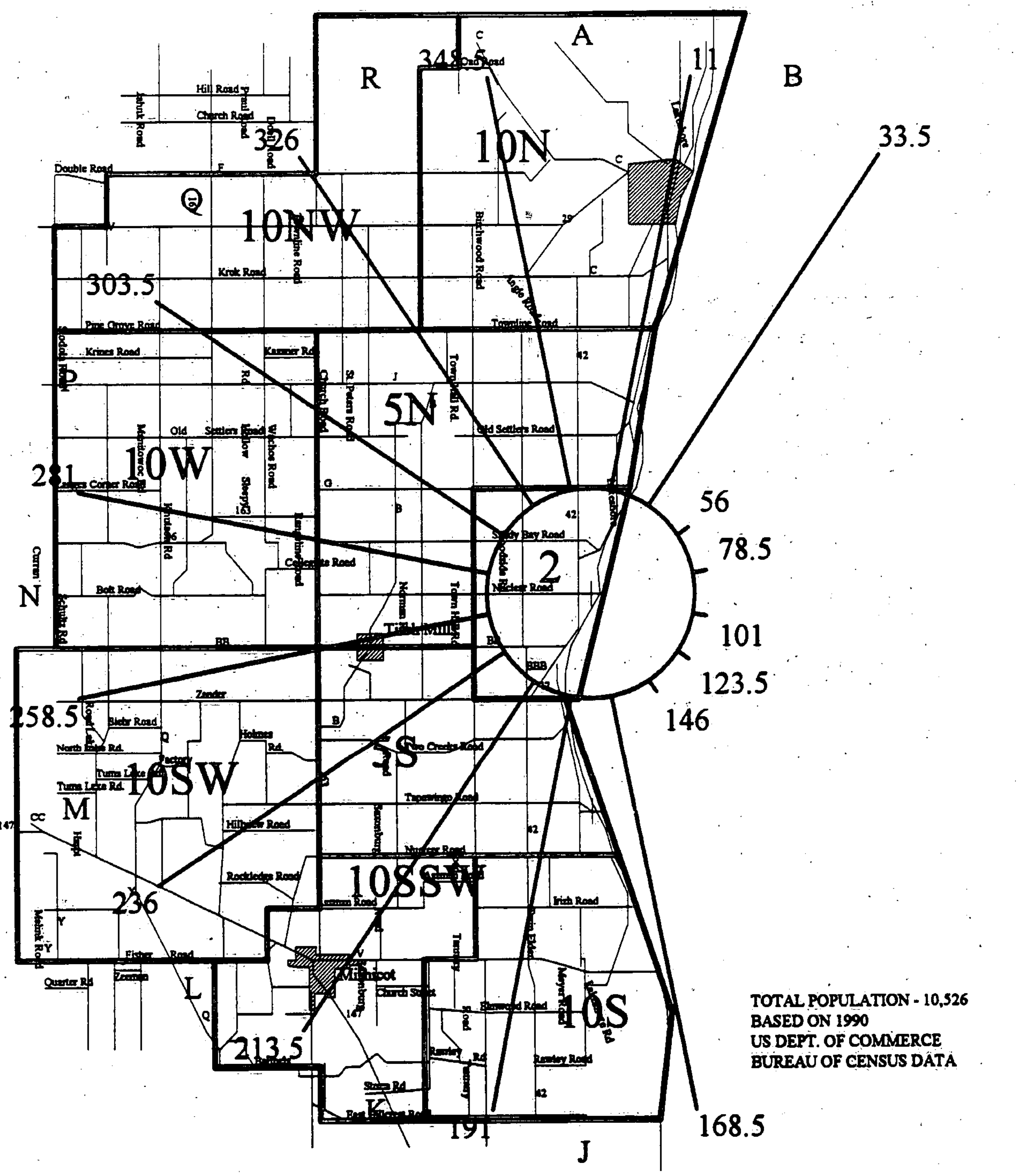


FIGURE C-5

EPZ GRID MAPS ARE LOCATED IN THE FOLLOWING LOCATIONS:

1)Technical Support Center

2)Emergency Operations Facility

3)Site Access Facility

4)Radiation Protection Office/Radiological Analysis Facility

APPENDIX D

Index of Letters of Agreement

TITLE	DATE WRITTEN	LOCATION
I. FEDERAL		
A. U.S. Nuclear Regulatory Commission	June, 1987 (NUREG 0728)	Emergency Plan Files
B. U.S. Department of Energy	September 13, 1994	Emergency Plan Files
II. STATE		
A. Department of Military Affairs - Division of Emergency Government	June 6, 1994	Emergency Plan Files
1. The following agencies will respond as directed by the Department of Military Affairs a. Department of Agriculture b. Department of Health and Social Services, Division of Health c. Department of Administration d. Department of Natural Resources, Division of Enforcement c. Department of Transportation, Division of Highways f. Department of Transportation, Division of Enforcement and Inspection, State Patrol		
B. University of Wisconsin - Hospital and Clinics	January 10, 1995	Emergency Plan Files
C. University of Wisconsin - Milwaukee	November 30, 1994	Emergency Plan Files
III. COUNTY		
A. Kewaunee County DEG	September 7, 1994	Emergency Plan Files
B. Kewaunee County Sheriff's Department	July 27, 1994	Emergency Plan Files
C. Manitowoc County DEG	September 2, 1994	Emergency Plan Files
D. Manitowoc County Sheriff's Department	July 28, 1994	Emergency Plan Files

TITLE	DATE WRITTEN	LOCATION
IV. CITY/TOWN		
A. Ambulanee Service		
1. Kewaunee Ambulance Service 2. Mishicot Area Ambulance Service 3. Two Rivers Fire Department Ambulance	~~April 3, 1995~~ (Extended through December 31, 1996 November 1, 1995 April 3, 1995	Emergency Plan Files Emergency Plan Files Emergency Plan Files
B. City of Kewaunee Fire Department	Contract (Annual Renewal)	Purchasing Files
V. PRIVATE ORGANIZATIONS		
A. American Nuclear Insurers	Contract (Annual renewal)	Insurance Files
B. Institute of Nuclear Power Operations	January 31, 1996	Emergency Plan Files
C. Nuclear Support Services	Contract (December 31, 1996)	Purchasing Files
D. Teledyne - Brown Engineering Midwest Laboratory	April 14, 1994	Emergency Plan Files
E. Two Rivers Community Hospital	March 3, 1995	Emergency Plan Files
F. *Westinghouse Electric Corporation	December 22, 1994 (Response Plan)	Emergency Plan Files
G. Wisconsin Electric Power Company	November 29, 1994	Emergency Plan Files
H. Ameritech (Wisconsin Bell)	March 1, 1995	Emergency Plan Files
I. WPSC Lakeshore Division	March 21, 1994	Emergency Plan Files
J. WPSC Kewaunee District	March 11, 1996	Emergency Plan Files

* Perpetual letter established

APPENDIX E

Radiological Emergency Response Plans

1. STATE OF WISCONSIN RADIOLOGICAL EMERGENCY RESPONSE PLAN
2. KEWAUNEE COUNTY RADIOLOGICAL EMERGENCY RESPONSE PLAN
3. MANITOWOC COUNTY RADIOLOGICAL EMERGENCY RESPONSE PLAN

(Copies are located in the Technical Support Center, Emergency Operations Facility, and the offices of the Nuclear Emergency Preparedness Supervisor.)

APPENDIX F

Emergency Equipment, Supplies and Reference Materials

INDEX

See EPMP 10.1 for inventory requirements.

TECHNICAL SUPPORT CENTER

ITEM

Emergency Plan

Emergency Plan Implementing Procedures

State of Wisconsin Radiological Incident Response Plan (Vol. I and II)

Kewaunee County Emergency Plan (Vol. II of State Plan)

Manitowoc County Emergency Plan (Vol. II of State Plan)

Updated Safety Analysis Report

Technical Specifications

Operating Procedures

Plant Drawing Aperture Card

10-Mile EPZ, Sector/Grid Map

Potassium Iodide

X/Q Meteorological Overlays

Beta Air Monitor

Portable Air Sampler/Filters (Available in adjacent RAF)

Portable Radiation Monitor (Available in adjacent RAF)

Computer Terminal with Access to Plant Process Computer

See EPMP 10.1 for inventory procedure.

CONTROL ROOM

ITEM

Emergency Plan -

Emergency Plan Implementing Procedures

Potassium Iodide

Portable Radiation Monitor

10-Mile EPZ, Sector/Grid Map

See RT-SAE-83 for inventory procedure.

HEALTH PHYSICS

<u>ITEM</u>

<u>Radiation Protection Office</u>

Emergency Plan

Emergency Plan Implementing Procedures

High Band Portable Radios and Chargers

<u>Radiological Analysis Facility</u>

Emergency Plan

Emergency Plan Implementing Procedures

Radio Remote Console

High Band Portable Radios

10-Mile EPZ Sector/Grid Map

X/Q Meterological Overlays

See RC-HP-115 for inventory procedure.

OPERATIONAL SUPPORT FACILITY

ITEM

Emergency Plan -

Emergency Plan Implementing Procedures

Portable Radiation Monitor (Available in adjacent RAF)

See PMP-83-1 for inventory procedure.

EMERGENCY OPERATIONS FACILITY

ITEM

Emergency Plan

Emergency Plan Implementing Procedures

Technical Specifications

Updated Safety Analysis Report

Operating Procedures

Nuclear Emergency Public Information Plan

Domestic Drawing Micro-Film Aperture Card Library
(Covering: Structural, Mechanical, Electrical)

State of Wisconsin Radiological Incident Response Plan (Vol. I and II)

Kewaunee County Emergency Plan (Vol. II of State Plan)

Manitowoc County Emergency Plan (Vol. II of State Plan)

50-mile EPZ Ingestion Pathway Map

10-mile EPZ Sector/Grid Map

See EPMP 10.1 for inventory procedure.

SITE BOUNDARY FACILITY

ITEM

Emergency Plan -

Emergency Plan Implementing Procedures

Protective Clothing

Dosimetry Equipment

Portable Radiation Monitor

Environmental Monitoring Team Kits

Communication Equipment

See HPF-115 for Inventory Checklist.

SECURITY BUILDING

ITEM

Emergency Plan

Emergency Plan Implementing Procedures

See Security Procedure, SCP 30.2

APPENDIX G

List of EPIPs and Cross-References to the Emergency Plan

Emergency Plan Implementing Procedures			Emergency Plan Section
A. Administrative Procedures			
1.	EP-AD-1	Plant Personnel Response to an Emergency	6.4.1
2.	EP-AD-2	Emergency Class Determination	4.1, 4.1.1, 4.2
3.	EP-AD-3	Unusual Event	4.1.1
4.	EP-AD-4	Alert, Site Emergency or General Emergency	4.1.2, 4.1.3, 4.1.4
5.	EP-AD-5	Emergency Response Organization Shift Relief Guideline	
6.	EP-AD-6	Deleted	
7.	EP-AD-7	Emergency Notifications from KNPP	4.1.1, 6.1
8.	EP-AD-8	Deleted	
9.	EP-AD-9	Deleted	
10.	EP-AD-10	Deleted	
11.	EP-AD-11	Emergency Radiation Controls	6.5.1, 6.5.2
12.	EP-AD-12	Deleted	
13.	EP-AD-13	Deleted	
14.	EP-AD-14	Deleted	
15.	EP-AD-15	Recovery Planning and Termination	9.0, 9.1, 9.2, 9.3, 9.4
16.	EP-AD-16	Occupational Injuries or Vehicle Accidents During Emergencies	6.5.2, 6.5.3, 6.5.4

Emergency Plan Implementing Procedures		Emergency Plan Section
17. EP-AD-17	Deleted	
18. EP-AD-18	Potassium Iodide Distribution	6.5.3
19. EP-AD-19	Protective Action Guidelines	6.4.2

B. Emergency Operations Facility Procedures

1. EP-EOF-1	Deleted	
2. EP-EOF-2	Emergency Operations Facility (EOF) Activation	4.1.2, 7.1.5
3. EP-EOF-3	Corporate Action for Unusual Event	4.1.1, 4.1.4, 6.1
4. EP-EOF-4	Corporate Action for Alert, Site Emergency or General Emergency	4.1.2, 4.1.3, 4.1.4
5. EP-EOF-5	Deleted	
6. EP-EOF-6	Deleted	
7. EP-EOF-7	Deleted	
8. EP-EOF-8	Notification of Alert, Site Emergency or General Emergency	4.1.2, 4.1.3, 4.1.4, 6.1
9. EP-EOF-9	Deleted	
10. EP-EOF-10	Deleted	
11. EP-EOF-11	Communcations Documentation	7.2
12. EP-EOF-12	Media Center/Emergency Operation Facility/Joint Public Information Center Security	5.22

Emergency Plan Implementing Procedures			**Emergency Plan Section**
C. Operational Procedures			
1.	EP-OP-1	Deleted	
2.	EP-OP-2	Deleted	
3.	EP-OP-3	Deleted	
D. Operational Support Facility Procedures			
1.	EP-OSF-1	Deleted	
2.	EP-OSF-2	Operational Support Facility Operations	4.1.2, 7.1.4
3.	EP-OSF-3	Work Requests During an Emergency	5.2.2, 6.3
4.	EP-OSF-4	Search and Rescue	6.4.1
E. Security Procedures			
1.	EP-SEC-1	Deleted	
2.	EP-SEC-2	Security Force Response to Emergencies	5.2.2
3.	EP-SEC-3	Security Force Response to Personnel Accountability	5.2.2, 6.4.1
4.	EP-SEC-4	Security Force Actions for Dosimetry Issue	5.2.2, 6.7.1
5.	EP-SEC-5	Security Force Response to Personnel Evacuation	6.4.1
F. Radiation Emergency Team Procedures			
1.	EP-RET-1	Deleted	
2.	EP-RET.2	In-Plant Radiation Emergency Team	1.3, 4.1.2, 5.2.2, 6.2.1, 6.6.1
3.	EP-RET.2A	RPO-RAF Activation	4.1.2, 7.1.3

Emergency Plan Implementing Procedures			Emergency Plan Section
4.	EP-RET-2B	Gaseous Effluent Sample and Analysis	6.2.2, 6.2.3
5.	EP-RET-2D	Emergency Radiation Entry, Controls and Implementation	6.6.1, 6.7.1, 6.7.2
6.	EP-RET-2E	Deleted	
7.	EP-RET-2F	Deleted	
8.	EP-RET-3	Emergency Chemistry Team	1.3, 5.2.2, 6.2.2
9.	EP-RET-3A	Liquid Effluent Release Paths	6.2.2
10.	EP-RET-3B	Deleted	
11.	EP-RET-3C	Post Accident Operation of the High Radiation Sample Room	7.3.1
12.	EP-RET-3D	Containment Air Sampling Analysis Using CASP	7.3.1
13.	EP-RET-3E	Deleted	
14.	EP-RET-4	Site Radiation Emergency Team	1.3, 5.2.2, 6.6.1
15.	EP-RET-4A	SAF Operation/Relocation	6.6.1, 6.7.2, 7.1.9
16.	EP-RET-4B	Deleted	
17.	EP-RET-4C	Deleted	
18.	EP-RET-4D	SAM-II Operation	6.2.2, 6.2.4
19.	EP-RET-5	Site Boundary Dose Rate During Controlled Plant Cooldown	6.2.3, 7.3.2(1)
20.	EP-RET-5A	Deleted	
21.	EP-RET-6	Deleted	
22.	EP-RET-7	Deleted	

Emergency Plan Implementing Procedures		Emergency Plan Section
23. EP-RET-8	Contamination Control at the Two Rivers Community Hospital	6.7.4
24. EP-RET-9	Post-Accident Population Dose	7.2.3

G. Environmental Procedures

1. EP-ENV-1	Environmental Monitoring Group Organization and Responsibilities	1.3, 5.2, 5.2.2, 6.2.4 Appendix A
2. EP-ENV-2	SAF Activation for Environmental Monitoring Teams	4.1.2, 7.1.7
3. EP-ENV-3A	Deleted	
4. EP-ENV-3B	Deleted	
5. EP-ENV-3C	Dose Projection Using KRDose Software	6.2.3, 7.3.2(1)
6. EP-ENV-3D	Revision and Control of KRDose	
7. EP-ENV-3E	Deleted	
8. EP-ENV-3F	Deleted	
9. EP-ENV-3G	Deleted	
10. EP-ENV-3H	Deleted	
11. EP-ENV-4A	Portable Survey Instrument Use	6.2.4, 7.3.2(2)
12. EP-ENV-4B	Air Sampling Analysis	6.2.4, 7.3.2(2)
13. EP-ENV-4C	Environmental Sampling Techniques	6.2.4, 7.3.2(2)
14. EP-ENV-4D	Plume Tracking for Environmental Monitoring Teams	6.2.4, 7.3.2(2)
15. EP-ENV-4E	Deleted	
16. EP-ENV-5A	Deleted	

Emergency Plan Implementing Procedures			Emergency Plan Section
17.	EP-ENV-5B	Deleted	
18.	EP-ENV-5C	Deleted	
19.	EP-ENV-5D	Deleted	
20.	EP-ENV-5E	Deleted	
21.	EP-ENV-6	Deleted	
22.	EP-ENV-8	Deleted	

H. Technical Support Center Procedures

1.	EP-TSC-1	Technical Support Center Organization and Responsibilities	5.2, 5.2.2, Appendix A
2.	EP-TSC-2	Technical Support Center Activation	4.1.2, 7.1.2
3.	EP-TSC-3	Plant Status Procedure	6.2.1, 6.3, 7.1.2
4.	EP-TSC-4	Emergency Design Change, Major Equipment Repair	6.3
5.	EP-TSC-5	Deleted	
6.	EP-TSC-6	Deleted	
7.	EP-TSC-7	RV Head Venting Time Calculation	7.3.1
8.	EP-TSC-8A	Calculations for Steam Release from Steam Generators	7.3.1
9.	EP-TSC-8B	STMRLS Computer Program	7.3.1
10.	EP-TSC-9A	Core Damage Assessment	7.3.1
11.	EP-TSC-9B	Core Computer Program	7.3.1
12.	EP-TSC-10	Technical Support for IPEOP's	

Emergency Plan Implementing Procedures	Emergency Plan Section
I. Appendices	
A. Communications	7.2, 7.2.1
B. Forms	7.2.2

ATTACHMENT 3

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 12, 1996

Letters of Concurrence from the State of Wisconsin and Both Kewaunee and Manitowoc Counties Supporting Emergency Action Level Changes Based on the Acceptable Deviations Described in EPPOS No. 1, Implemented in Revision 18.

PUBLIC SERVICE

WISCONSIN PUBLIC SERVICE CORPORATION

April 8, 1996

Mr. Paul Schmidt
Section of Radiation Protection
Division of Health
P.O. Box 309
Madison, WI 53701-0309

Dear Mr. Schmidt:

The Federal Code of Regulations (Part 50, App. E, Section B) requires an annual review of utility Emergency Action Levels (EAL's) by State and local authorities. This year's review is different from previous years in that this revision proposes the elimination of various EAL's which were based on exceeding plant Technical Specifications or events that by themselves would not lead to a reactor core damaging event. Because of the proposed removal of these EAL's, the NRC requires a written statement of support for these changes from both State and local authorities.

Attached to this letter are the following documents:

1. A draft copy of the proposed changes to Emergency Plan Implementing Procedure, EP-AD-2 (Rev. U), "Emergency Class Determination."

2. A NRC document, Subject: Acceptable Deviations from Appendix 1 to NUREG-0654 Based Upon the Regulatory Analysis of NUMARC/NESP-7, "Methodology for Development of Emergency Action Levels." (NOTE: A copy of NUMARC/NESP-7 is available upon request.)

3. A draft copy of the KNPP safety evaluation to be submitted to the NRC along with this revision.

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If you have any questions or comments regarding the content or format of this revision to the KNPP Emergency Action Levels (EAL's) , please feel free to contact me so clarification can be provided.

Upon completion of your review, please sign and date the Review and Concurrence box at the end of this letter and return the entire letter to me.

Your prompt review would be appreciated. Thank you for your support.

Sincerely,

David R. Seebart
David R. Seebart
Nuclear Emergency Preparedness Supervisor

jlm

cc (w/o attach.): Division of Emergency Government
Mr. K.H. Evers, Wisconsin Public Service Corporation
Mr. R.P. Pulec, Wisconsin Public Service Corporation

REVIEW AND CONCURRENCE

The WI Radiation Protection Unit
(Organization)
has reviewed and concurs with the EAL changes proposed by the Kewaunee Nuclear Power Plant as stated in Procedure EP-AD-2 (Rev. U), "Emergency Class Determination."

Paul Schmidt 4-15-96
Signature Date

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PUBLIC SERVICE

WISCONSIN PUBLIC SERVICE CORPORATION

April 8, 1996

Ms. Nancy Crowley, Director
Manitowoc County Emergency Government
1025 South Ninth Street
Manitowoc, WI 54220

Dear Ms. Crowley:

The Federal Code of Regulations (Part 50, App. E, Section B) requires an annual review of utility Emergency Action Levels (EAL's) by State and local authorities. This year's review is different from previous years in that this revision proposes the elimination of various EAL's which were based on exceeding plant Technical Specifications or events that by themselves would not lead to a reactor core damaging event. Because of the proposed removal of these EAL's, the NRC requires a written statement of support for these changes from both State and local authorities.

Attached to this letter are the following documents:

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3. A draft copy of the KNPP safety evaluation to be submitted to the NRC along with this revision.

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Kewaunee Nuclear Power Plant • North 490, Hwy.42 • Kewaunee, WI 54216-9510 • (414) 388-2560

If you have any questions or comments regarding the content or format of this revision to the KNPP Emergency Action Levels (EAL's), please feel free to contact me so clarification can be provided.

Upon completion of your review, please sign and date the Review and Concurrence box at the end of this letter and return the entire letter to me.

Your prompt review would be appreciated. Thank you for your support.

Sincerely,

David R. Seebart

David R. Seebart
Nuclear Emergency Preparedness Supervisor

jlm

cc (w/o attach.): Division of Emergency Government
Mr. K.H. Evers, Wisconsin Public Service Corporation
Mr. R.P. Pulec, Wisconsin Public Service Corporation

REVIEW AND CONCURRENCE

~~The~~ Manitowoc County Emergency Management
(Organization)
has reviewed ~~and concurs with~~ the EAL changes proposed by the Kewaunee Nuclear Power Plant as stated in Procedure EP-AD-2 (Rev. U), "Emergency Class Determination."

Nancy H. Crowley 4-19-96
Signature Date

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5/18/96
Lyle did not send the first page back to me CRS

Mr. Lyle Schmiling
Kewaunee County Emergency Government

April 8, 1996

If you have any questions or comments regarding the content or format of this revision to the KNPP Emergency Action Levels (EAL's), please feel free to contact me so clarification can be provided.

Upon completion of your review, please sign and date the Review and Concurrence box at the end of this letter and return the entire letter to me.

Your prompt review would be appreciated. Thank you for your support.

Sincerely,

David R Seebart

David R. Seebart
Nuclear Emergency Preparedness Supervisor

jlm

cc (w/o attach.): Division of Emergency Government
Mr. K.H. Evers, Wisconsin Public Service Corporation
Mr. R.P. Pulec, Wisconsin Public Service Corporation

REVIEW AND CONCURRENCE

The KEWAUNEE COUNTY EMERGENCY GOVERNMENT
(Organization)
has reviewed and concurs with the EAL changes proposed by the Kewaunee Nuclear Power Plant as stated in Procedure EP-AD-2 (Rev. U), "Emergency Class Determination."

Lyle E Schmiling 4-23-96
Signature Date

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ATTACHMENT 4

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 12, 1996

Kewaunee Nuclear Power Plant Technical Specification Amendments No. 104 and No. 118 that Apply to Emergency Action Levels Implemented in Revision 18.

K-93-245
12/13/93

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 9, 1993

Docket No. 50-305

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54037-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. M86417)

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications in response to your application dated May 4, 1993.

The amendment modifies the KNPP Technical Specifications in accordance with Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP)," dated January 31, 1989.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Richard J. Laufer

Richard J. Laufer, Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 104 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

NRC LETTER DISTRIBUTION

T A Hanson (MG&E)
M W Seitz (WPL)
Larry Nielsen (ANFC)
D A Bollom G6
D E Cole KNP
K H Evers KNP
J P Giesler KNP
K A Hoops KNP
M L Marchi KNP
D L Masarik KNP
J N Morrison D1
L A Nuthals (NSRAC)
R P Pulec D2 (2)
C A Schrock D2
C S Smoker KNP
C R Steinhardt D2
C A Sternitky KNP
T J Webb KNP
S F Wozniak D2
QA Vault KNP

9312140395 2pp

cc:

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service
  Corporation
Post Office Box 19002
Green Bay, Wisconsin 54037-9002

Foley & Lardner
Attention: Mr. Bradley D. Jackson
One South Pinckney Street
P. O. Box 1497
Madison, Wisconsin 53701-1497

Chairman
Town of Carlton
Route 1
Kewaunee, Wisconsin 54216

Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

Chairman
Public Service Commission of
Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Attorney General
114 East, State Capitol
Madison, Wisconsin 53702

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, Wisconsin 54216

Regional Administrator - Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Robert S. Cullen
Chief Engineer
Wisconsin Public Service Commission
P. O. Box 7854
Madison, Wisconsin 53707

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated May 4, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

9312140401 18pp.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 104, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: December 9, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS i	TS i
TS iv	TS iv
TS v	TS v
TS vi	TS vi
TS vii	-----
TS 1.0-5	TS 1.0-5
TS 1.0-6	TS 1.0-6
TS 1.0-7	TS 1.0-7
TS 6.9-3	TS 6.9-3
TS 6.9-4	--------
TS 6.9-5	--------
TS 6.9-6	--------
TS 6.10-2	TS 6.10-2
TS 6.16-1	TS 6.16-1
---------	TS 6.16-2
---------	TS 6.16-3
TS 6.17-1	TS 6.17-1
TS 6.18-1	TS 6.18-1
TS Section 7/8	----------

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TECHHICAL SPECIFICATIONS
APPENDIX A

LIST OF TABLES*

*Text shifted to different page

LIST OF FIGURES *

*Text shifted to different page

m. <u>RATED POWER</u>

RATED POWER is the steady-state reactor core output of 1,650 MWt.

n. <u>REPORTABLE EVENT</u>

A REPORTABLE EVENT is defined as any of those conditions specified in 10 CFR 50.73.

o. <u>RADIOLOGICAL EFFLUENTS</u>

1. MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

2. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by TS 6.16.b, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by TS 6.9.b.1 and TS 6.9.b.2.

3. PROCESS CONTROL PROGRAM (PCP)

The PCP shall contain the current formulae, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, federal and state regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

4. SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

5. UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

p. STANDARD SHUTDOWN SEQUENCE

When a LIMITING CONDITION FOR OPERATION is not met, and a plant shutdown is required except as provided in the associated action requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 36 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the specified time limits as measured from the time of determination of the failure to meet the LIMITING CONDITION FOR OPERATION. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable when the plant is in COLD or REFUELING SHUTDOWN.

q. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (µCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0361	I-132
0.2703	I-133
0.0169	I-134
0.0838	I-135

'ext shifted to Page TS 1.0.6

Amendment No. ~~64~~,~~75~~,~~100~~,~~102~~,104

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the ODCM and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

6. Records of transient or operational cycles for these facility components.

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of meetings of the NSRAC and PORC.

10. Records for Environmental Qualification.

11. Records of reviews performed for changes made to the ODCM and the PCP.

6.16 RADIOLOGICAL EFFLUENTS

a. Written procedures shall be established, implemented and maintained covering the activities referenced below:

1. PCP implementation.

2. ODCM implementation.

3. Quality Assurance Program for effluent and environmental monitoring.

b. The following programs shall be established, implemented, and maintained:

1. <u>Radioactive Effluent Controls Program</u>

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

(a) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM.

(b) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2.

(c) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM.

(d) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50.

(e) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days.

(f) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2% of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

(g) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1.

(h) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

(i) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

(j) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

2. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurement of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

(a) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

(b) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

(c) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.17 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. The documentation shall contain:

   (a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   (b) A determination that the change will maintain the overall conformance of the soldified waste product to existing requirements of federal, state, or other applicable regulations.

2. Shall become effective upon review and acceptance by the PORC.

6.18 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

a. The ODCM shall be approved by the Commission prior to implementation.

b. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by TS 6.10.b.11. This documentation shall contain:

   (a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   (b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the PORC.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. The date the changes were made shall be indicated. In addition, a method such as redlining should be used to clearly identify the changes.

SECTION 7/8 AND ALL SECTION 7/8 TABLES HAVE BEEN DELETED

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated May 4, 1993, the Wisconsin Public Service Corporation (WPSC), the licensee, submitted a request for revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications. The proposed amendment would implement revisions to the KNPP TS identified by the NRC's Generic Letter (GL) 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP)."

Specifically, the changes to implement GL 89-0I would:

1. Incorporate programmatic controls in the Administrative Controls section of the TS that satisfy the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a and Appendix I to 10 CFR Part 50.

2. Relocate the existing procedural details in current specifications involving radioactive effluent monitoring instrumentation, the control of liquid and gaseous effluents, equipment requirements for liquid and gaseous effluents, radiological environmental monitoring, and radiological reporting details from the TS to the ODCM.

3. Relocate the definition of solidification and existing procedural details in the current specification on solid radioactive wastes to the PCP.

4. Simplify the associated reporting requirements.

5. Simplify the administrative controls for changes to the ODCM and PCP.

6. Add record retention requirements for changes to the ODCM and PCP.

7. Update the definitions of the ODCM and PCP consistent with these changes.

~~9312140408~~ 4pp

2.0 EVALUATION

On January 3I, I989, the staff issued GL 89-01. In this GL, the staff noted that it had examined the contents of the RETS in relation to the Commission's Interim Policy Statement of Technical Specifications Improvements and had determined that programmatic controls could be implemented in the Administrative Controls section of the TS to satisfy the existing regulatory requirements for RETS. The staff had also determined that the procedural details of the TS on radioactive effluents and radiological environmental monitoring could be relocated to the ODCM, while the procedural details for solid radioactive waste could be relocated to the PCP. These procedural details are not required to be included in the TS by 10 CFR 50.36a. After relocation, future changes to these procedural details will be controlled by the controls for changes to the ODCM and PCP included in the Administrative Controls section of the TS.

In the GL, the staff provided model specifications and encouraged licensees to propose changes consistent with the GL. The licensee's proposed changes to the Kewaunee TS are in accordance with the guidance provided in GL 89-01 and are addressed below.

(1) The licensee has proposed to incorporate programmatic controls for radioactive effluents and radiological environmental monitoring in Specification 6.16, "Radiological Effluents," of the TS as noted in the guidance provided in GL 89-0I. The programmatic controls ensure that programs are established, implemented, and maintained to ensure that operating procedures are provided to control radioactive effluents consistent with the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50.

(2) The licensee has confirmed that the detailed procedural requirements addressing Limiting Conditions for Operation, their applicability, remedial actions, associated surveillance requirements, or reporting requirements for TS Section 7/8 have been relocated to the ODCM or PCP, as appropriate. These changes to the ODCM and PCP have been prepared in accordance with the proposed changes to TS 6.17 and TS 6.I8, and meet the specified criteria. The procedural details that have been removed from the TS are not roquired by the Commission's regulations to be included in the TS. The RETS, as relocated to the ODCM and PCP, can be subsequently changed by the licensee in accordance with 10 CFR 50.59 without prior NRC approval. As stated in new TS 6.10.b.11, the licensee's records of reviews performed for changes made to the ODCM and PCP will be retained for the duration of the operating license.

(3) The licensee has proposed replacing the existing specifications in the Administrative Controls section of the TS for the Annual Radiological Environmental Operating Report (TS 6.9.b.1), for the Semiannual Radioactive Effluent Release Report (TS 6.9.b.2), for the PCP (TS 6.17), and for the ODCM (TS 6.18), with the updated specifications that were provided in GL 89-01, with some editorial changes. Existing reporting details of TS 6.9.b.1 and TS 6.9.b.2 have been relocated to the ODCM.

In addition TS 6.9.b.2.A.(3), "Solid Waste Shipped," has been relocated to the PCP.

(4) TS Definitions 1.0.o.1, 1.0.o.5, 1.0.o.9, 1.0.o.10, and 1.0.o.11; the definitions of Gaseous Radwaste Treatment System, Purge-Purging, Ventilation Exhaust Treatment System, Venting, and Radiological Environmental Monitoring Manual, respectively, were proposed for deletion and relocation to the ODCM, consistent with the deletion and relocation to the ODCM of the sections that refer to them. Although these specific changes were not listed in GL 89-01, they are consistent with the intent of the GL, and are reflective of the nonstandard nature of the licensee's TS. Remaining definitions in TS Section 1.0.o were renumbered to maintain the numbering consistency of the TS.

(5) Renumbered TS Definitions 1.0.o.2 and 1.0.o.3, the definitions of ODCM and PCP, respectively, have been proposed for updating consistent with the guidance of GL 89-01 to reflect their change in scope.

(6) Definition 1.0.o.7, Solidification, was proposed for deletion from the TS and relocation to the PCP, consistent with the guidance of GL 89-01.

On the basis of the above, the staff finds that the changes included in the proposed TS amendment are consistent with the guidance provided in GL 89-01. 8ecause the control of radioactive effluents continues to be limited in accordance with operating procedures that must satisfy the regulatory requirements 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, the staff concludes that this change is administrative in nature and there is no adverse impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there have been no public comments on such finding (58 FR 39062). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. King
R. Laufer

Date: December 9, 1993

K-95-55

Rec'd. 4-24-95

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 17, 1995

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M90879)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 118 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications (TS) in response to your application dated November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995.

The amendment revises the KNPP TS 3.1.d, "Leakage of Reactor Coolant," TS 4.2.b, "Steam Generator Tubes," and TS 3.4.a, "Steam Generators," to allow application of a voltage-based repair limit for the steam generator (SG) tube support plate (TSP) intersections experiencing outside diameter stress corrosion cracking (ODSCC). The amendment also reduces the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day (gpd) to 150 gpd. These changes to the tube repair criteria are applicable for the 1995 to 1996 operating cycle (Cycle 21) only.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 118 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

NRC to WPSC LETTER DISTRIBUTION

T A Hanson (MG&E)	K H Evers KNP	C S Smoker KNP
M W Seitz (WPL)	M L Marchi D2	C R Steinhardt D2
Larry Nielsen (ANFC)	L A Nuthals (NSRAC)	CA Sternitzky KNP(Lic)
D A Bollom G6	R P Pulec KNP (3)	S F Wozniak D2
D E Day D1	C A Schrock KNP	B J Domnick D2 (Com)

9504240159 2pp

Mr. M. L. Marchi
Wisconsin Public Service Corporation   Kewaunee Nuclear Power Plant

cc:

Foley & Lardner
Attention: Mr. Bradley D. Jackson
One South Pinckney Street
P. O. Box 1497
Madison, Wisconsin 53701-1497

Chairman
Town of Carlton
Route 1
Kewaunee, Wisconsin 54216

Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

Chairman
Public Service Commission of
Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Attorney General
114 East, State Capitol
Madison, Wisconsin 53702

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, Wisconsin 54216

Regional Administrator - Region III
U. S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4531

Mr. Robert S. Cullen
Chief Engineer
Wisconsin Public Service Commission
P. O. Box 7854
Madison, Wisconsin 53707

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

9504240763 18pp.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.118 , are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: April 17, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS ii	TS ii
TS 3.1-9	TS 3.1-9
TS B3.1-10	TS B3.1-10
TS B3.1-11	TS B3.1-11
TS B3.1-12	TS B3.1-12
TS 3.4-1	TS 3.4-1
TS 3.4-3	TS B3.4-1
TS 3.4-4	TS B3.4-2
TS 3.4-5	---------
TS 4.2-3	TS 4.2-3
TS 4.2-4	TS 4.2-4
TS 4.2-5	TS 4.2-5
TS 4.2-6	TS 4.2-6
--------	TS 4.2-7
--------	TS 4.2-8
TS B4.2-4	TS B4.2-4

d. Leakage of Reactor Coolant

1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 500 gallons per day through any one steam generator except when the tube support plate, voltage-based repair criteria is applied. Primary to secondary leakage is limited to 150 gallons per day through any one steam generator when the tube support plate voltage-based repair criteria is applied. With tube leakage greater than the above limit, reduce the leakage rate within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.

4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.

5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is operable.

Leakage of Reactor Coolant (TS 3.1.d)[18]

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}\ \mu Ci/cc$ ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}\ sec/m^3$, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.5 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve (12) hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

The 150 gpd leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

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[18] USAR Sections 6.5, 11.2.3, 14.2.4

TS 3.1.d.3

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.

TS 3.1.d.4

The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.

TS 3.1.d.5

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

   If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[19]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[20]. Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range. The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[21][22]

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

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[19] USAR Section 4.2

[20] USAR Section 9.2

[21] USAR Table 3.2-1

[22] USAR Figure 3.2-8

3.4 STEAM AND POWER CONVERSION SYSTEM

APPLICABILITY

Applies to the OPERATING status of the Steam and Power Conversion System.

OBJECTIVE

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

SPECIFICATION

a. Steam Generators

1. The reactor shall not be heated > 350°F unless the following conditions are satisfied.

   A. Two steam generators are OPERABLE.

      1. System piping and valves directly associated with providing auxiliary feedwater flow to the steam generators are OPERABLE.

      2. Five main steam safety valves per OPERABLE steam generator are OPERABLE, except during required surveillance tests or during in-service testing of these valves and steam generators in accordance with 10 CFR 50.55a, provided that at least two main steam safety valves associated with the steam generator under test are OPERABLE.

   B. A minimum of 39,000 gallons of water is available in the condensate storage tanks and the Service Water System is capable of delivering an unlimited supply from Lake Michigan.

   C. The DOSE EQUIVALENT 1-131 on the secondary side of the steam generators does not exceed 0.1 μCi/cc.

2. If, when the reactor is > 350°F, any one of the conditions of TS 3.4.a.1 cannot be met within 48 hours, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System < 350°F within an additional 12 hours

BASIS

Steam Generators (TS 3.4.a)

Two steam generators are required to be OPERABLE when the average reactor coolant temperature is > 350°F to ensure that sufficient heat removal capability exists for power operation and decay heat removal. Although one steam generator would provide sufficient decay heat removal capability, two steam generators are required in order to provide the necessary redundancy to meet the single failure criterion. An OPERABLE steam generator is defined by TS 3.4.a.

The ten main steam safety valves (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr at 1181 lbs. pressure. The maximum full-power steam flow at 1721 MWTH is 7,449,000 lbs./hr; therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary. The requirement that five main steam safety valves per OPERABLE steam generator are available will assure sufficient steam relief capability.

Testing of the main steam system while the plant is in HOT SHUTDOWN conditions is permitted provided that at least two main steam safety valves associated with the steam generator under test are available to provide sufficient relief capacity to protect the system during the test.

The specified minimum water supply in the condensate storage tanks is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement. When AC power is available, unlimited replenishment of the condensate storage supply is available from Lake Michigan through the Service Water System.

An evaluation was performed to determine the maximum permissible steam generator primary-to-secondary leak rate during a steam line break event. The evaluation considered both a preaccident and accident initiated iodine spike. The results of the evaluation show that the accident initiated spike yields the limiting leak rate. This evaluation was based on a 30 REM thyroid dose at the site boundary and initial primary and secondary coolant iodine activity levels of 1.0 μCi/gm and 0.1 μCi/gm DOSE EQUIVALENT I-131 respectively. A leak rate of 34.0 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the steam generator faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm, the standard operating leakage limit applied for the tube support plate voltage-based plugging criteria specified in TS 4.2.b.5.

Auxiliary Feedwater Pumps (TS 3.4.b)

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned to both steam generators; the discharge of the turbine-driven pump, which starts automatically, is aligned to backup both motor-driven pumps. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

It is acceptable to exceed 350°F with an inoperable turbine-driven auxiliary feedwater pump. However, operability of the pump must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

With no auxiliary feedwater pumps OPERABLE, action shall be taken to restore a pump as soon as possible. The action with three pumps inoperable is to maintain the plant in an operating condition in which the auxiliary feedwater system is not needed for heat removal. When one pump is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no auxiliary feedwater pumps available, there would be no feedwater to the steam generator to cool the plant to 350°F when the Residual Heat Removal System could be placed in operation.

Turbine Overspeed Protection System (TS 3.4.c)

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the redundant overspeed trip system (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two-out-of-three channels to trip.

REFERENCES

USAR Section 10
USAR Section 14.1

1. <u>Steam Generator Sample Selection and Inspection</u>

The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. <u>Steam Generator Tube Sample Selection and Inspection</u>

The tubes selected for each in-service inspection shall:

a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 3% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted below and in TS 4.2.b.2.b.

Tubes left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

Implementation of the steam generator tube support plate voltage-based plugging criteria requires a 100% bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

d. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories, and actions taken as described in Table 4.2-2.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes, and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTE: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service inspection, result in all inspection results falling into the C-1 category; or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.2-2 fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.a.1.C or,

   2. A seismic occurrence greater than the Operating Basis Earthquake, or

   3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

   4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

4. Plugging Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied.(2)

a. Any tube which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Tube repair shall be in accordance with the methods described in WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)" or CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves."

b. Any Westinghouse mechanical sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged prior to returning the steam generator to service. Figure TS 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

(2)The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

5. Tube Support Plate Voltage-Based Plugging Criteria[3]

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage $\leq$ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but $\leq$ 5.6 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > 5.6 volts will be plugged or repaired.

d. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of TS 4.2.b.5.c. If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

[3]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

Where:

V = measured voltage
V_{BOC} = voltage at BOC
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 9.6 volt for 7/8 inch tubes

6. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degradation.

3. Identification of tubes plugged.

4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with Specification 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines the tube support plate.

4. If the calculated conditional burst probability exceeds the threshold value, notify the NRC and provide an assessment of the safety significance of the occurrence.

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

<u>Technical Specification 4.2.b.5</u>[5]

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

<u>Technical Specification 4.2.b.6</u>

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).

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[5]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise the KNPP TS 3.1.d, "Leakage of Reactor Coolant," TS 4.2.b, "Steam Generator Tubes," and TS 3.4.a, "Steam Generators," to permit the use of a voltage-based steam generator tube repair criteria for defects confined to within the thickness of the tube support plate. The amendment would also reduce the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day (gpd) to 150 gpd. All of the proposed changes to the tube repair criteria would be applicable for the 1995 to 1996 operating cycle (Cycle 21) only.

The proposed voltage-based tube repair criteria pertain specifically to outside diameter stress corrosion cracking (ODSCC) flaws. The proposed criteria would: (1) permit flaws confined to within the thickness of the tube support plate with bobbin voltages less than or equal to 2.0 volt to remain in service; (2) permit flaws confined to within the thickness of the tube support plate with bobbin voltages greater than 2.0 volt but less than or equal to 5.6 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation; and (3) require flaw indications confined to within the thickness of the tube support plate with bobbin voltages greater than 5.6 volts to be plugged or repaired.

Additional clarifying information with respect to implementation of the voltage-based tube repair criteria was provided in the licensee's letters dated January 9, February 14, March 8, and April 3, 1995.

2.0 BACKGROUND

The NRC staff is currently developing a generic interim position on voltage-based limits for ODSCC confined to within the thickness of the tube support plates. The staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," and in a draft generic letter titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the Federal Register on August 12, 1994. However, the staff is continuing to evaluate an acceptable generic position which will take into consideration public comments on the draft generic letter cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants. The staff currently plans to document its final position on this matter in a generic letter. Pending completion and issuance of the staff's final generic position on the voltage-based tube repair criteria, the staff is continuing to evaluate voltage-based repair criteria proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety. Furthermore, these case-specific evaluations limit the applicability of the voltage-based repair criteria to one cycle of operation.

In a letter dated November 8, 1994, the licensee requested an amendment to modify the technical specifications to allow the use of a voltage-based steam generator tube repair criteria. Based on subsequent discussions between the licensee and the NRC staff, the licensee provided a revised amendment request by letter dated March 8, 1995, which modified the request to apply only to Cycle 21 and provided clarifying information. Additional clarifying information was also provided in letters dated January 9, February 14, and April 3, 1995.

The tube repair limits proposed by the licensee include a lower voltage repair limit of 2.0 volts for axially oriented ODSCC flaws confined to within the thickness of the tube support plates in lieu of the present criteria which is a depth-based limit of 40% or 50% depending on the degradation mechanism. In addition, the repair limits allow bobbin indications between 2.0 and 5.6 volts (the upper voltage repair limit) to remain in service provided inspection of these indications with a RPC probe does not confirm the degradation to be present.

The licensee's proposal is similar to that reviewed and approved for several other plants and has been reviewed on a case-specific basis. The tube structural limit is based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. The margin of safety of 3 against burst during normal operation is inherently satisfied since the structural constraint provided by the tube support plates, which surround the degradation to which the voltage-based repair criteria applies, ensures these tubes will maintain this margin of safety at these locations. To complement these deterministic criteria, the conditional probability of burst under accident conditions and the primary-to-secondary leakage from the steam generator tubes during a postulated main steam line break (MSLB) are also calculated.

3.0 PROPOSED INTERIM TUBE REPAIR CRITERIA

Kewaunee Technical Specifications 3.1.d, 4.2.b.2, 4.2.b.4, 4.2.b.5, and 4.2.b.6 and Bases 3.1.d, 4.2.b.5, and 4.2.b.6, would be revised by this proposed amendment to specify the tube repair and leakage criteria for ODSCC confined to within the thickness of the tube support plate. The proposed changes to the tube repair and leakage criteria in the technical specifications specify, in part:

a. Implementation of the steam generator tube support plate voltage-based plugging criteria requires a 100% bobbin coil probe inspection for all hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known ODSCC indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least 20% random sampling of tubes inspected over their full length.

b. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage ≤ 2.0 volts will be allowed to remain in service.

c. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in Item (d) below.

d. Indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but ≤ 5.6 volts may remain in service if a RPC inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage > 5.6 volts will be plugged or repaired.

e. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersections or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of Item (d) above. If the bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}\left(V_{SL} - V_{BOC}\right) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

where:

V = bobbin voltage
V_{BOC} = voltage at the beginning of cycle (BOC)
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where the operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 9.6 volts for 7/8-inch tubes

f. For implementation of the voltage-based repair criteria to tube support plate intersections, notification of the NRC staff prior to returning the steam generators to service is required should any of the following conditions arise:

   (1) If the estimated leakage based on the actual measured end-of-cycle (EOC) voltage distribution would have exceeded the leak limit (for the postulated MSLB using licensing basis assumptions) during the previous operating cycle.

   (2) If circumferential crack-like indications are detected at the tube support plate intersections.

   (3) If indications are identified that extend beyond the confines of the tube support plate.

   (4) If the calculated conditional burst probability exceeds the threshold value. Additionally, an assessment of the safety significance of this occurrence should be provided.

g. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited, in part, to 150 gpd through any one steam generator when the tube support plate voltage-based repair criteria is applied.

In addition to the above proposed technical specification changes, the licensee also made the following commitments for implementing the voltage-based repair criteria:

1. All bobbin indications with voltages greater than 1.5 volts will be inspected with a RPC probe. RPC probe inspections assist in identifying axial ODSCC as the dominant mechanism for indications at the tube support plates.

2. Tubes with bobbin dent voltages exceeding 5.0 volts, large mixed residual, or indications of copper deposits will be inspected with a RPC and any RPC flaw indications detected at these intersections will be dispositioned in accordance with the depth-based repair criteria.

3. Tubes with known leaks will be repaired prior to returning the steam generators to service.

4. Steam generator tube integrity data (i.e., voltage distributions and leak/burst evaluations) will be provided to the NRC within 90 days following restart.

5. A 0.720-inch diameter bobbin coil probe will be used during the steam generator inspections at intersections where the voltage-based repair criteria will be applied.

6. The NRC will be notified prior to plant restart if any primary water stress corrosion cracking (PWSCC) indications are detected within the tube support plate intersections during the steam generator inspection. Additionally, the eddy current analysts will be briefed on the potential that PWSCC can occur at the tube support plate locations.

7. The conditional probability of burst and the primary-to-secondary leakage calculation will be performed in accordance with the guidance provided in the draft generic letter using the methodology described in WCAP-14277.

8. The conditional probability of burst calculation will be compared against a threshold value of $1x10^{-2}$.

In general, the licensee intends to follow the guidance of the draft generic letter with the following exceptions: (1) calibration of the bobbin coil probe on the 4-20% through-wall holes rather than the 4-100% through-wall holes; (2) implementation of the probe wear standard; (3) limiting new probe variability; (4) removing specimens for destructive examination; and (5) the application of data exclusion criteria. These exceptions are discussed below.

4.0 <u>EVALUATION</u>

4.1 <u>Inspection Issues</u>

In support of the proposed voltage-based repair limits, the licensee proposes to utilize the eddy current test guidelines included as Appendix A to WCAP-12985, Revision 2, dated March 1993, and as later supplemented. The inspection criteria are intended to ensure the inspection scope, data acquisition, and data analysis are performed in a manner consistent with the methodology utilized to develop the voltage limits. The proposed guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators.

The inspections to be performed as part of the voltage-based repair criteria include both bobbin coil and rotating pancake coil (RPC) examinations. Bobbin coil examinations will be performed for 100% of the hot-leg tube support plate intersections and cold-leg intersections down to the lowest cold-leg tube support plate with known ODSCC. The determination of the tube support plate having ODSCC indications will be based on a minimum 20% random sampling of the tubes over their full length. The bobbin coil examinations for intersections at which the voltage-based repair criteria will be applied will be performed with a 0.720-inch bobbin coil probe. RPC examinations will be performed to permit additional characterization of the flaws found with the bobbin coil probe and to inspect intersections with significant bobbin interference signals (due to copper deposits, dents, large mix residuals) which may impair the ability of the bobbin coil probe to detect flaws or which may unduly influence the bobbin voltage measurement.

With respect to flaw characterization, a key purpose of the RPC inspections is to ensure the absence of detectable crack-like circumferential indications and detectable indications extending outside the thickness of the tube support plate. The voltage-based repair criteria are not applicable to intersections exhibiting such indications (i.e., circumferential indications and indications extending outside the tube support plates), and special reporting requirements pertaining to the finding of such indications have been proposed if these types of indications are detected. RPC examinations will be performed (1) at all intersections with bobbin coil indications exceeding 1.5 volts, (2) at all intersections where the dent signal is greater than 5.0 volts as measured with the bobbin coil probe, (3) at intersections where the mixed residual could cause a 1.0 volt bobbin signal to be missed or misread (i.e., masked), and (4) at all intersections where copper deposits influence the bobbin coil signal. Any flaw-like indications found at intersections with dent signals greater than 5.0 volts, with large mixed residuals, or where copper deposits influence the bobbin coil signal will be dispositioned in accordance with the depth-based tube repair criteria.

As previously mentioned, tube support plate locations with bobbin dent voltages above 5.0 volts, as measured by the bobbin coil probe, will be inspected with an RPC probe. Inspections of dented intersections are performed, in part, as a result of (1) the possible masking effect the dent may have on the detection of flaw indications, (2) the possible development of primary water stress corrosion cracking (PWSCC) flaws at these locations, and (3) the possible development of circumferential cracks at these locations. With respect to masking flaw indications, it is anticipated that flaw signals on the order of 1.0 volt would have phase angles that fall within the flaw reporting range even if the bobbin dent voltage was as high as 5.0 volts based on a vectorial combination of the eddy current signals attributed to the flaw and to the dent. As a result, RPC inspecting all intersections with bobbin dent voltages in excess of 5.0 volts provides reasonable assurance that any structurally significant ODSCC indications will be detected and repaired. With respect to the occurrence of circumferential cracking at the support plate elevations, the RPC sampling plan provides assurance that if a significant amount of circumferential cracking is occurring at the tube support plate elevations it will be detected.

With respect to the occurrence of PWSCC at dented tube support plate intersections, the potential exists for axial PWSCC to occur at intersections where the bobbin dent voltage is less than 5.0 volts. Most frequently these types of indications (i.e., indications representative of axially oriented PWSCC) have been found at tube support plates with significant denting, have been known to occur at 180° spacing as two axial indications due to the stresses in the tube, and have been known to occur within the tube support plate but occasionally extending outside the tube support plate. Axial PWSCC is not presently analyzed as part of the voltage-based repair criteria. As a result of this and the potential for PWSCC to occur at dented intersections less than 5.0 volts, the licensee has proposed to (1) RPC inspect all bobbin indications which are greater than 1.5 volt at dented intersections (2) RPC inspect all intersections where the bobbin dent voltage is greater than 5.0 volts regardless of whether a bobbin indication is detected, and (3) notify the NRC prior to plant restart if any PWSCC indications are detected at the support plate elevations. In addition, the licensee will brief the eddy current analysts on the potential for PWSCC at tube support plate locations and the analysts will be instructed to report occurrences of axial PWSCC. The staff finds this sampling plan adequate to detect the onset of axial PWSCC at support plate locations. The staff also notes that frequently axial PWSCC extends outside the tube support plate intersection, making it more likely to be detectable with the bobbin coil. This provides added confidence that if extensive axial PWSCC is present, it will be detected. The staff notes that if PWSCC is detected at support plate elevations, an evaluation to ensure the voltage-based repair criteria is only applied to ODSCC indications will need to be performed and reviewed by the staff.

With respect to data acquisition and analysis, the licensee's eddy current guidelines either contain requirements or guidance pertaining to (1) recording all indications regardless of voltage amplitude, (2) controlling probe wear by the use of a probe wear standard, (3) calibrating the bobbin coil probes, and (4) using a transfer standard to ensure consistency between the voltages measured in the field and the voltages measured in the laboratory as part of the development of the voltage-based approach.

The staff notes that there are several outstanding technical issues with respect to the inspection guidelines, as documented in previously issued NRC documents (e.g., in draft NUREG-1477 and in the draft generic letter cited above) which will be resolved prior to issuing the final generic letter on voltage-based limits for ODSCC confined to within the thickness of the tube support plate. These outstanding issues include, in part, (1) limits on new probe variability, (2) the need to reinspect all tubes since the last successful probe wear check, (3) the need to calibrate the bobbin coil on the 4-100% holes versus the 4-20% holes, and (4) the capabilities/limitations of the 1-coil, 2-coil, and 3-coil RPC probes. However, the staff concludes that the inspection guidelines submitted by the licensee are acceptable since the proposed repair criteria is limited to one cycle, and the calibration, recording, and analysis requirements are consistent with the methodology used in the development of the tube repair criteria described in the draft generic letter.

4.2 Tube Integrity Issues

The thin-walled tubing of the steam generator constitutes more than half of the reactor coolant pressure boundary (RCPB), and maintenance of the structural and leakage integrity of this boundary is a requirement under Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), Appendix A. Specific requirements governing the maintenance of steam generator tube integrity are contained in the plant technical specifications and Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). These include requirements for periodic inservice inspection of the tubing, flaw acceptance criteria (i.e., repair limits for plugging or sleeving), and primary-to-secondary leakage limits. These requirements, coupled with the broad scope of plant operational and maintenance programs, have formed the basis for assuring adequate steam generator tube integrity.

Flaw acceptance criteria, termed plugging/repair limits, are specified in the plant technical specifications. The purpose of the technical specification repair limits is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria 14, 15, 30, 31 and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits.

The traditional strategy for accomplishing the objectives of the General Design Criteria related to steam generator tube integrity has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowances for eddy current measurement error and flaw growth between inspections have been added to the minimum wall thickness requirements, consistent with Regulatory Guide 1.121, to arrive at a depth-based repair limit. Development of the minimum wall thickness requirements to satisfy Regulatory Guide 1.121 was governed by analyses for uniform thinning of the tube wall in the axial and circumferential directions. The assumption of uniform thinning conservatively bounds the degrading effects of all flaw types currently occurring in the field and is the basis of the standard 40% depth-based repair limit incorporated into the technical specifications. However, the 40% repair limit is conservative for highly localized flaws such as pits and short cracks. In particular, the 40% depth-based repair limit is conservative for ODSCC that occurs at the tube support plate intersections.

Enforcement of a minimum wall thickness requirement for the steam generator tubes would implicitly serve to ensure leakage integrity during normal operation and postulated accidents, as well as structural integrity. It has been recognized, however, that defects, especially cracks, may occasionally grow entirely through-wall and develop small leaks. For this reason, limits on the allowable primary-to-secondary leakage have been established in a plant's technical specifications to ensure timely plant shutdown before adequate structural and leakage integrity of an affected steam generator tube is impaired.

The proposed voltage-based tube repair limits consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criteria represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from examination of steam generator tubes removed from a number of steam generators in operating nuclear power plants shows that for bobbin indications exceeding 2.0 volts (i.e., the lower voltage repair limit), maximum crack depths range between 50% and 100% through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For indications at or near 5.6 volts (i.e., the upper voltage repair limit), the maximum crack depths have been found to generally range between 90% and 100% through-wall. Many of the tubes which will be allowed to remain in service under the proposed voltage-based repair criteria may have or develop through-wall or near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents.

The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.3 and 4.4 of this evaluation.

Although the voltage-based repair limits ensure adequate structural and leakage integrity, the NRC staff recognizes that overall margins have been reduced when compared to the margins associated with the existing 40% depth-based repair limit. Because of the increased likelihood of through-wall cracks developing in service, the staff has included provisions for augmented steam generator inspections, as discussed in the previous section, and more restrictive operational tube leakage limits, as discussed below.

4.3 Structural Integrity

4.3.1 Deterministic Structural Integrity Assessment

The licensee has proposed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 2.0 volt lower voltage repair limit would retain adequate structural margins, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled steam generator tube data from other plants (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized and are consistent with the calibration standard voltage set-ups and voltage measurement procedures to be used by the licensee during the steam generator inspections.

To confirm the nature of the degradation occurring at the tube support plate elevations, the licensee pulled three tubes with five tube support plate intersections from the steam generators during an outage in the Spring of 1993. Tube pulls confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC and also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). Destructive examination of these tube support plate intersections was performed. The examinations performed confirmed that the dominant degradation mechanism for the indications at the support plate elevations was axially oriented ODSCC and that the voltage-based tube repair criteria for indications at the tube support plates was applicable at Kewaunee.

The draft generic letter on voltage-based repair criteria provides guidance on performing tube pulls for initial implementation of the repair criteria. In summary, the draft generic letter states that at least six tube support plate intersections should be obtained either during the outage in which the voltage-based repair criteria is implemented or during the inspection outage preceding initial application of the voltage-based repair criteria. To follow the draft generic letter guidance on tube pulls, the licensee would need to pull 6 intersections from their steam generators during this outage since their last tube pulls were two outages ago. The current guidance in the draft generic letter on the issue of tube pulls gives no consideration to the length of the operating interval between inspections and can result in plants with short operating intervals removing more tubes (in the long run) than a plant with longer operating intervals. As a result of this and other public comments received on this issue, the staff has been evaluating alternative options to the tube pull guidance in the draft generic letter. The latest guidance was presented to the industry during a public meeting on January 18, 1995. The licensee believes their tube pulls met the intent of this guidance as discussed in a letter from the licensee dated February 14, 1995 and, as a result, the licensee does not intend to pull tubes during the upcoming outage. Pending finalization of the generic letter position on tube pulls, the staff has concluded that the licensee need not remove tubes during the upcoming outage to meet the guidance in the draft generic letter.

The voltage-based tube repair criteria previously approved by the staff for other plants have been set deterministically to ensure that indications accepted for continued service with this repair criteria will retain adequate structural integrity during the full range of normal, transient, and postulated accident conditions. The repair criteria includes allowances for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Because the voltage-based repair criteria addresses tubes affected with ODSCC confined to within the thickness of the tube support plates during normal operation, the staff has concluded that the structural constraint provided by the tube support plates ensures that all tubes to which the voltage-based criteria applies will retain a margin of 3 with respect to burst under normal operating conditions, consistent with the criteria of Regulatory Guide 1.121. For a postulated MSLB accident, however, the tube support plates may displace axially during blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plates. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

The allowable end-of-cycle (EOC) voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions (i.e., 3660 psi), in accordance with Regulatory Guide 1.121, is based on the lower 95% prediction interval of the burst pressure/bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95/95 confidence level. This voltage limit is approximately 9 volts for the 7/8-inch diameter tubing used in the Kewaunee steam generators. The difference between the 9 volt allowable EOC voltage and the 2.0 volt repair criterion represents an allowance of approximately 7 volts for voltage growth (i.e., ODSCC flaw growth) during the forthcoming fuel cycle (i.e., Cycle 21) and for eddy current voltage measurement variability (i.e., the repeatability error) during the steam generator inspection.

To demonstrate the adequacy of the voltage-based repair criteria, the largest RPC confirmed indication which may be left in service (i.e., a 2.0 volt indication), was analyzed by the staff to determine if the indication would grow to the point that the structural voltage limit (i.e., approximately 9 volts) is exceeded. In this analysis, a 2.0 volt bobbin indication is assumed to grow at a rate equal to the maximum growth rate observed during the latest cycle for which data is available (i.e., 1.24 volts for Cycle 19 which was 0.89 effective full power years (EFPY) in duration) and it is assumed that the indication was undersized by 20% (i.e., the 95% cumulative probability of the non-destructive examination (NDE) uncertainty). The resultant EOC voltage is determined from this analysis to be 4.2 volts for the 1.3 EFPY planned for Cycle 21. This EOC voltage compares favorably to the structural voltage limit determined from the burst pressure versus bobbin voltage correlation.

The proposed 2.0 volt lower repair limit is applicable to all bobbin indications confirmed by RPC or which have not been RPC inspected. The licensee is also proposing a 5.6 volt upper voltage repair limit applicable to bobbin indications which have been RPC inspected but for which the RPC failed to confirm the bobbin indication. This 5.6 volt upper voltage repair limit can be derived from the information in EPRI Report TR-100407, Revision 1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," dated August, 1993. The maximum voltage which ensures a margin of 1.43 with respect to burst under postulated MSLB conditions (i.e., 3660 psi) for tubes with lower bound material properties at a 95% prediction interval was 9.6 volts based on the data available at that time. A 5.6 volt upper voltage repair limit was calculated from the 9.6 volt structural limit by including an allowance for average growth rates of 50% of the BOC voltage amplitude and an allowance of 20% for eddy current voltage measurement variability (i.e., the 95% cumulative probability of the NDE uncertainty).

Since the issuance of EPRI Report TR-100407, Revision 1 in August 1993, additional data has been added to the burst pressure database used in the development of this upper voltage repair limit and several of the existing data points in the database have been updated as a result of additional analysis. However, taking this into consideration with the growth rates and the planned operating interval for Kewaunee, the staff has concluded that the 5.6 volt upper voltage repair limit is adequate for this cycle of operation. The new upper voltage repair limit was calculated to be approximately 5.6 volts for Kewaunee assuming an allowance of approximately 40% for flaw/voltage growth over the next operating cycle (i.e., Cycle 21) and an allowance of 20% for measurement variability. The voltage measurement variability estimate considers measurement variabilities stemming from bobbin coil probe wear and variability in the analysts' interpretation of the bobbin coil voltage. Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during prior cycles at Kewaunee. Over the last few cycles (typically between 0.8 and 1.0 EFPYs), the average percent voltage growth at Kewaunee has been 18% (1991 to 1992), 5% (1992 to 1993), and 13% (1993 to 1994). The 40% average growth allowance used to support the approximately 5.6 volt upper voltage repair limit is intended to provide margins for variation in future growth rates at Kewaunee and for the increased length of the operating interval (i.e., 1.3 EFPY). As a result of the above analysis, the staff concludes that the 5.6 volt upper voltage repair limit is acceptable for Kewaunee.

The staff has evaluated the acceptability of the upper voltage repair limit for indications below this limit which may be left in service if detected by the bobbin coil probe but not confirmed to be flaw-like by the RPC probe. Short and/or relatively shallow cracks detected by the bobbin coil may sometimes not be detectable by the RPC probe, although the RPC probe is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance. Furthermore, the burst strength of steam generator tubing affected by predominantly axially oriented ODSCC at the support plate elevations is not a unique function of the bobbin voltage. Rather, for a given voltage, there is a statistical distribution of possible burst strengths, as indicated in the burst pressure/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the RPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). That is, ODSCC which is not detectable by RPC is believed to be less likely to affect the tube structural and leakage integrity during the operating cycle than ODSCC which is detectable by both the bobbin coil and the RPC probe. In addition, the burst and leakage potential for bobbin indications accepted for continued service under the 5.6 volt criterion have been directly considered in the probability of burst and leakage assessments described below, with no credit given to the fact that RPC failed to confirm the indications. Based on these considerations, the staff finds the upper voltage repair limit of 5.6 volts for indications which may be left in service if detected by bobbin inspection but not confirmed by the RPC to be acceptable.

4.3.2 <u>Probabilistic Structural Integrity Assessment</u>

A probabilistic analysis for the potential for steam generator tube ruptures, given a MSLB, must also be performed. The need for this analysis, which supplements the deterministic analysis discussed above, is dictated by the following considerations:

1. The deterministic analysis does not consider the tail of the burst pressure distribution beyond the lower 95% prediction interval used to determine the maximum allowable EOC voltage. Given the large numbers of indications which could potentially be accepted for continued service with the 2.0 volt criterion, the probabilistic analysis ensures that the use of the 95% prediction interval value in lieu of the 99% or 99.9% values does not lead to a significant likelihood of steam generator tube rupture given a MSLB.

2. The deterministic assessment ignores the burst and leakage potential of bobbin indications between 2.0 volt and 5.6 volts for which the RPC probe failed to confirm the indication. The probabilistic assessment, however, considers the burst potential of these indications with no credit given for the lack of confirmation by the RPC probe of the presence of these indications.

3. The deterministic analysis does not account for bobbin indications missed by the data analysts. The staff concluded in draft NUREG-1477 and in the draft generic letter that the probabilistic assessment is required in order to address the burst potential of indications missed by the data analysts.

4. The deterministic analysis does not consider the cumulative effect of the entire distribution of indications accepted for continued service.

Employing the probabilistic analysis, however, ensures that all indications accepted for continued service are accounted for in determining the overall probability of burst given a MSLB.

5. The deterministic analysis does not consider the tails of the material properties distribution and the eddy current voltage variability distributions. The probabilistic analysis does include the entire distribution of material properties and voltage variability.

To perform the probabilistic analysis, the EOC distribution of indications must be determined. Consistent with the approach recommended in the draft generic letter on voltage-based repair criteria, the BOC distribution used in the determination of the EOC distribution involves adjusting the indications detected during the inspection by the probability of detection (POD), where the POD is assumed to have a constant value of 0.6, irrespective of voltage. The net effect of this assumption is that the distribution of detected bobbin indications is scaled up by a factor of 1/POD. After this POD scaling is made, indications removed from service by tube repair (i.e., plugging or sleeving) are subtracted from this distribution to yield the assumed BOC distribution. The EOC distribution is then determined by combining the voltage measurement uncertainty distribution, the voltage growth rate distribution, and the BOC voltage distribution using Monte Carlo techniques. For each of the resultant EOC voltages determined by the above analysis, the distribution of burst pressures as a function of bobbin voltage along with a distribution of material properties is sampled by Monte Carlo techniques to yield a distribution of burst pressures for the EOC voltage distribution. The conditional probability of burst, given a MSLB, can then be determined by dividing the number of times the Monte Carlo analysis yields a burst pressure below the MSLB differential pressure for the EOC voltage distribution by the total number of samples. A distribution of material tensile properties is sampled in the probabilistic analysis since the data points in the bobbin voltage/burst pressure correlation have been normalized to a flow stress of 75 ksi.

The POD scaling approach cited above is reasonably consistent with reported operating experience to-date with ODSCC in terms of accounting for the projected distribution of indications at EOC which were not previously detectable at BOC. However, operating experience to-date, for ODSCC confined to within the thickness of the tube support plate, is that maximum EOC bobbin voltages generally do not exceed 4 or 5 volts. Although there are known cases where indications on the order of 3 volts have not been detected, there is very little experience regarding the likelihood of not detecting bobbin indications between 3 and 10 volts. The industry believes that the numerical value of the POD is substantially higher than 0.6 for indications exceeding 1.0 volt, based, in part, on data collected from the Electric Power Research Institute (EPRI) performance demonstration program. However, pending further staff review, the staff believes a POD value of 0.6 is appropriate for this voltage-based repair criteria application.

The licensee will perform the probabilistic analysis discussed above which assumes the degradation is free span and ignores the potential constraining effects of the tube support plates. In addition, this analysis will be performed in a manner which considers the uncertainty in the parameters for the supporting correlations (e.g., burst pressure/bobbin voltage correlation). The results of the probabilistic analysis will be compared to a threshold value established by the staff. Consistent with the draft generic letter this threshold value is 1×10^{-2}. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844. In addition, the threshold value provides an indication that one or more tubes may not maintain the Regulatory Guide 1.121 safety margins for the entire operating cycle. The licensee has stated that the results of the probability of burst analysis will be compared against a threshold value of 1×10^{-2}. If this threshold value is exceeded, the NRC staff will be notified and an assessment of the safety significance of this occurrence will be provided to the NRC staff prior to returning the steam generators to service. The staff notes that all applicable data should be included in the burst pressure database when performing this calculation, except as discussed below.

4.3.3 <u>Data Exclusion from the Burst Pressure Correlation</u>

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in a database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the bobbin voltage/burst pressure database was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the correlations used in the bobbin voltage/burst pressure database in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the burst pressure/bobbin voltage database. The specific criteria are presented in a letter referenced by the licensee which was submitted to the NRC by the Electric Power Research Institute (EPRI) on April 22, 1994.

The data points excluded from the burst pressure/bobbin voltage database as a result of applying these criteria are listed in Table E-1 of the subject document. The staff has concluded that excluding the data points listed in Table E-1 from the 7/8-inch diameter steam generator tubing burst pressure/bobbin voltage database is appropriate since it meets the exclusion criteria discussed by the staff at the February 8, 1994, industry meeting. Pending further evaluation of the generic criteria presented in Section E.2, the staff is continuing to assess the appropriateness of excluding data points from the burst pressure correlation on a case-by-case basis.

4.3.4 Combined Accident Loadings

Combined accident condition loadings such as loss of coolant accident (LOCA) plus safe shutdown earthquake (SSE) could result in yielding at a tube support plate (TSP) with subsequent deformation of the tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could propagate through-wall resulting in the potential for in-leakage under LOCA conditions. In-leakage is a potential concern as leakage through several severed tubes may inhibit the core refill/reflood process and cause an unacceptable increase in the core peak clad temperature (PCT).

The most limiting accident conditions from tube deformation considerations are seismic (SSE) plus LOCA. The seismic excitation applied to steam generators is defined in the form of acceleration response spectra at the steam generator supports. In the seismic analysis, the licensee has used generic response spectra, which envelope the Kewaunee specific response spectra. A finite element model of the Series 51 steam generator was developed and the analysis was performed using the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam, and pipe elements as well as general matrix input to represent the piping and support stiffnesses. Interactions at the TSP/shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

LOCA loads developed as a result of transient flow following a postulated primary coolant pipe break were calculated for five different pipe break locations. These included three large and two minor pipe breaks. The large pipe break locations evaluated were the steam generator inlet and outlet lines and the reactor coolant pump outlet line, while the minor pipe breaks analyzed were the pressurizer surge line and the accumulator line breaks. Prior qualification of the Kewaunee primary piping for leak before break requirements resulted in the limiting LOCA event being either the accumulator line break or the pressurizer surge line break. The licensee has however, used the loads for the primary piping break as a conservative approximation.

The principal tube loading from a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the tube U-bends. A differential pressure is created across the two legs of the tube, which causes an inplane horizontal motion of the U-bends and induces significant lateral loads on the tube. The pressure time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave induced loadings, the predominant motion of the U-bends is along the plane of the U-bend. Thus, the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two TSPs.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the steam generator caused by the break hydraulics and reactor coolant loop motion. However, the resulting TSP loads from this motion are small compared to those due to the rarefaction wave induced motion.

To obtain the LOCA induced hydraulic forcing functions, a dynamic blowdown analysis is performed to generate the system hydraulic forcing functions assuming an instantaneous double-ended guillotine break. The hydraulic forcing functions are then applied, along with the displacement time-history of the reactor pressure vessel (obtained from a separate reactor vessel blowdown analysis), to a system structural model, which includes the steam generator, the reactor coolant pump and the primary piping. This analysis yields the time history displacements of the steam generator at its upper lateral and lower support nodes. These time-history displacements formulate the forcing functions for obtaining the tube stresses due to LOCA shaking of the steam generator.

In calculating combined TSP loads, the LOCA rarefaction and LOCA shaking loads are combined directly, while the LOCA and SSE loads are combined using the square root of the sum of the squares. The overall TSP load is transferred to the steam generator shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding in the TSP at the wedge support. Some tubes in the vicinity of the wedge supports could partially deform and subsequently collapse during a LOCA. The reduction in flow area increases the resistance to flow of steam from the core, which in turn may potentially increase PCT. In addition, there is a potential concern that partial through-wall cracks in a steam generator tube could progress to through-wall cracks during tube deformation. The resulting in-leakage is a potential concern since the cumulative leakage may cause an increase in the core PCT.

Utilizing results from recent tests and analysis programs, the licensee has shown that tubes will undergo permanent deformation if the change in diameter exceeds 0.025-inch. This threshold for tube deformation is related to the concern for tubes with preexisting through-wall cracks that could potentially open during a combined LOCA plus SSE event. For the Kewaunee plant, the LOCA plus SSE loads were determined to be of such magnitude that none of the tubes are predicted to exceed this deformation limit and therefore, will not be subjected to significant tube leakage.

The licensee has assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously performed tests, the licensee has concluded that bending stresses on the order of yield stress of the tube material is necessary before the burst strength of the tube is affected to any significant degree. The maximum calculated bending stress in a tube wall during a seismic event is substantially less than the yield stress of the tube material. Thus, it is concluded that the burst strength of tubes with through-wall cracking is not affected by SSE event.

8ased on a review of the information provided by the licensee for the Kewaunee plant, it is concluded that no significant tube leakage is likely to occur during and SSE plus LOCA event, which has been identified as the most limiting condition from tube deformation considerations.

4.4 Leakage Integrity

An important implication of voltage-based steam generator tube repair criteria is that the criteria may permit tubes to have, or to develop, through-wall or near through-wall cracks during the forthcoming operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Thus, the leakage integrity of these tubes, in addition to their structural integrity, must be assessed.

The staff finds that adequate leakage integrity during normal operating conditions is reasonably assured by the technical specification limits on allowable primary-to-secondary leakage. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next operating cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded. The radiological consequences of this are discussed in Section 4.5.

4.4.1 Normal Operational Leakage

Implementation of the voltage-based tube repair criteria includes a reduction in the technical specification reactor coolant system leakage limits. Specifically, the present technical specification limit of 500 gallons per day (gpd) for primary-to-secondary leakage through any one steam generator is reduced to 150 gpd.

The present 500 gpd limit per steam generator is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of Regulatory Guide 1.121. Development of the 150 gpd per steam generator leakage limit has utilized the extensive industry database regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95% confidence interval for a given crack size, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length for MSLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length corresponding to a burst pressure of three times normal operating pressure.

The reduced steam generator leakage limits to be adopted for implementation of the voltage-based tube repair criteria are more restrictive than the present operating leakage limits in the plant's technical specifications in order to provide a margin of safety against rupture. This reduction in the steam generator maximum allowable leakage limits is also intended to provide an additional margin in the event that a crack grows at a rate much greater than expected or which may unexpectedly extend outside the thickness of the tube support plate. The staff finds the proposed operating leakage limits in technical specification 3.1.d.2 to be acceptable for implementation of the voltage-based tube repair criteria.

4.4.2 <u>Accident Leakage</u>

The licensee has proposed a model for calculating the steam generator tube leakage from the faulted steam generator during a postulated MSLB which consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage (POL) model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model).

In the POL model, the probability that a given indication will leak is presented as a function of the bobbin coil voltage of that indication. The data is separated into two categories (i.e., indications which leak during a MSLB and those which do not). While various functional forms can be fitted to the data, the staff has concluded that a single functional form, the log-logistic, is acceptable for the purpose of assessing MSLB-induced steam generator tube leakage. The staff believes that any non-conservatism associated with the use of the log-logistic model, as compared to the other functional forms, is small compared to the conservatism inherent in the existing methodology for calculating the steam generator tube leakage and the radiological consequences of this leakage induced by a postulated MSLB. In addition, the differences in the POL functional forms are considered to be less significant when the leakage is calculated using a linear leak rate model, as discussed below, instead of a constant leak rate model which treats leakage as independent of voltage.

Regarding the conditional leak rate model, a correlation between the steam generator tube leak rate and bobbin voltage data based on a linear regression fit of the logarithms of the data has been developed. The staff provided statistical criteria in the draft generic letter on voltage-based repair criteria which permits licensees to use such a correlation if the correlation can be statistically justified at a 95% confidence level (i.e., a p-value of 5%). The staff concludes that using a linear relationship between the logarithms of the leak rate and bobbin voltage is appropriate in the determination of the primary-to-secondary steam generator tube leakage during a postulated MSLB provided the statistical criteria delineated in the draft generic letter on this subject are met. If the statistical criteria in the draft generic letter are not met, the linear regression should be assumed to have zero slope (i.e., the linear regression fit should be assumed to be constant with voltage). The staff further notes that the databases used in such evaluations should be consistent with the databases discussed in Section 4.4.3 of this evaluation.

The licensee has proposed a method for determining the primary-to-secondary steam generator tube leakage during a postulated MSLB which involves a Monte Carlo method which simulates the regression parameter uncertainties. The staff has analyzed this model for the case where the p-value test is valid at the 5% level, and has concluded that this model is appropriate and consistent with the draft generic letter on voltage-based repair criteria. This method involves:

1. Determining random versions of the POL and leak rate correlations to account for the uncertainty in the regression parameters (i.e., parameter uncertainty).

2. Using the regression parameters from Step 1 to determine the leak rate for each flaw indication in the estimated EOC voltage distribution. The EOC voltage distribution used in this calculation is the same as that discussed in Section 4.3.2.

3. Calculating the sum of the individual leak rates determined in Step 2 to obtain a value of the total steam generator leak rate.

4. Repeating Steps 1, 2, and 3 many times (e.g., 10,000) to obtain a distribution of the total steam generator leak rates.

5. Ordering the distribution of total leak rates in Step 4 in ascending order, and taking the 95th quantile at a 95% confidence level as the primary-to-secondary steam generator leakage during a postulated MSLB. This is the value used in assessing the leakage integrity of the steam generator tubing.

The staff notes that some minor variations in the details of the modeling may be necessary for the case where the p-value test is invalid at the 5% level.

The licensee has calculated the allowable steam generator leak rate to be 34.0 gallons per minute (gpm) in the faulted steam generator. This value is intended to be consistent with maintaining the radiological consequences of a release outside containment to within a small fraction of the guideline values in 10 CFR Part 100 as discussed in Section 4.5. As a result, if the primary-to-secondary leakage during a postulated MSLB is less than the 34.0 gpm limit, steam generator tubing affected by axially oriented ODSCC at the tube support plate elevations will maintain adequate leakage integrity under these conditions. The staff, therefore, finds this limit acceptable.

4.4.3 Data Exclusion from the Leakage Correlations

During the performance of the pulled tube examinations, malfunctions in the test equipment or improper specimen preparation can occasionally occur which could result in erroneous readings. Data such as this should not be included in the database since it could result in invalid results and/or conclusions. The staff, therefore, concluded in draft NUREG-1477 that eliminating data from the conditional leak rate and probability of leakage databases was appropriate provided that the data could be shown to be erroneous or the result of an invalid test. The staff provided additional guidance regarding the exclusion of data from the databases used in the steam generator tube leakage evaluation in a meeting with the industry on February 8, 1994. As a result of this guidance, the industry provided criteria for determining whether data may be removed from the probability of leakage and conditional leak rate databases. The specific criteria are presented in a letter referenced by the licensee which was submitted to the NRC by the Electric Power Research Institute (EPRI) on April 22, 1994.

The data points excluded from the conditional leak rate database and the probability of leakage database as a result of applying these criteria are listed in Tables E-2 and E-3 of the EPRI April 22, 1994, letter. The staff has concluded that excluding the data points listed in Table E-2, with the exception of model boiler specimen 542-4 and pulled tube specimen J1-R8C74, from the 7/8-inch conditional leak rate database; and excluding the data points listed in Table E-3 from the 7/8-inch diameter POL database is appropriate since it meets the exclusion criteria discussed by the staff at the February 8, 1994, industry meeting. Pending further evaluation of the generic criteria presented in Section E.2 of the April 22, 1994 letter, the staff is continuing to assess the appropriateness of excluding data points from the conditional leak rate and POL database on a case-by-case basis.

4.5 Assessment of Radiological Consequences

In support of the amendment request, the licensee presented its assessment of the radiological dose consequences of a 34 gpm primary to secondary leak initiated by a main steam line break accident. In the assessment, the licensee assumed that the allowable activity level of dose equivalent ^{131}I was 1.0 μCi/g for the primary coolant and 0.1 μCi/g for the secondary coolant. Two assessments were presented. One was based upon a preexisting iodine spike and the other was based upon an accident initiated iodine spike. The licensee presented doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The licensee concluded that, based upon a limit of 30 rem thyroid at the EAB, a leak rate of 34 gpm was determined to be the upper limit for allowable primary to secondary leakage in the steam generator in the faulted loop.

The staff independently calculated the doses resulting from a main steamline break accident using the methodology associated with Standard Review Plan (SRP) 15.1.5, Appendix A. The assumptions which were utilized by the staff in its calculations are presented in the Attachment. The results of the staff's calculations confirm the licensee's conclusions that the doses would be less than the limits established by SRP 15.1.5, Appendix A.

5.0 SUMMARY

Based on the above evaluation, the staff concludes that adequate structural and leakage integrity of the indications accepted for continued service under the voltage-based repair criteria can be ensured for Cycle 21 (1995 to 1996) at Kewaunee, consistent with applicable regulatory requirements. The staff's approval of the proposed voltage-based repair criteria is based, in part, on the licensee being able to demonstrate that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 63127). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (I) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: K. Karwoski
J. Rajan
J. Hayes

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Attachment: Input Parameters for Kewaunee Evaluation of Main SteamLine Break Accident

INPUT PARAMETERS FOR KEWAUNEE EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 60 μCi/g of dose equivalent ^{131}I.

<u>Preexisting Spike Value (μCi/g)</u>

^{131}I = 32.7
^{132}I = 13.1
^{133}I = 53.8
^{134}I = 8.23
^{135}I = 29.5

2. Volume of primary coolant and secondary coolant.

Primary Coolant Volume (ft^3)	6236
Primary Coolant Temperature (°F)	578
Secondary Coolant Steam Volume (ft^3)	3838
Secondary Coolant Liquid Volume (ft^3)	1920
Secondary Coolant Steam Temperature (°F)	510.8
Secondary Coolant Feedwater Temperature (°F)	427.3

3. TS limits for DE ^{131}I in the primary and secondary coolant.

Primary Coolant DE ^{131}I concentration (μCi/g)	1.0
Secondary Coolant DE ^{131}I concentration (μCi/g)	0.1

4. TS value for the primary to secondary leak rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	150

5. Maximum primary to secondary leak rate to the faulted and intact SGs.

Faulted SG (gpm)	34
Intact SG (gpm)	0.1

6. Iodine Partition Factor

Faulted SG	1
Intact SG	0.1
Primary to Secondary Leakage	I.0

7. Steam Released to the environment

Faulted SG (lbs/2 hours)	99,300
Intact SG (lbs/2 hours)	209,000

8. Letdown Flow Rate (gpm) 40

9. Release Rate for 1 μCi/g of Dose Equivalent ^{131}I

		Ci/day
^{131}I	=	181
^{132}I	=	465
^{133}I	=	455
^{134}I	=	688
^{135}I	=	460

10. Atmospheric Dispersion Factors

EAB (0-2 hours)	2.9×10^{-4}
LPZ (0-8 hours)	5.2×10^{-5}