NRC-00-056

Public Service

Wisconsin Public Service Corporation
(a subsidiary of WPS Resources Corporation)
Kewaunee Nuclear Power Plant
North 490, Highway 42
Kewaunee, WI 54216-9511
920-388-2560

July 26, 2000

10 CFR 50, App. E

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Radiological Emergency Response Plan Implementing Procedures

Pursuant to 10 CFR 50 Appendix E, Wisconsin Public Service Corporation hereby submits one copy of the latest revisions to the Kewaunee Nuclear Power Plant Radiological Emergency Response Plan Implementing Procedures (EPIPs). These revised procedures supersede the previously submitted procedures.

Pursuant to 10 CFR 50.4, two additional copies of this letter and attachment are hereby submitted to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region III, Lisle, Illinois. As required, one copy of this letter and attachment is also submitted to the Kewaunee Nuclear Power Plant NRC Senior Resident Inspector.

Sincerely,

Mark L. Marchi
Vice President-Nuclear

DLF

Attachment

cc - US NRC Senior Resident Inspector, w/attach.
US NRC, Region III (2 copies), w/attach.
Electric Division, PSCW, w/o attach.
QA Vault, w/attach.

A045

KEWAUNEE NUCLEAR POWER PLANT

July 18, 2000

EMERGENCY PLAN IMPLEMENTING PROCEDURES TRANSMITTAL FORM

RETURN TO DIANE FENCL - KNPP

OUTSIDE AGENCY COPIES (1-20)

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T. Webb - NRC Region III (2 & 3)*
T. Webb - NRC Resident Inspector (4) (receives Appx. A phone numbers)*
T. Webb - State of Wisconsin (5)*
T. Webb - KNPP QA Vault w/NRC Letter (15)*
Bob Hayden – Wisconsin Electric Power Co. (10)
Craig Weiss – Wisconsin Power & Light (11)

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L. Renier-Hicks – GB-D2 Nuclear EOF (77)
J. Mueller - OSF (52)
C. Hutter - ATF-1 (64)
LOREB – ATF-1 (66)
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M. Anderson - CR/SS Office (51, 56)
L. Renier-Hicks – GB-D2 Nuclear (84)
J. Mueller - TSC (50)
C. Long - RAF (53)
C. Long - SBF/EMT (54)
C. Long - RPO (55)

WORKING COPIES (101-199) These copies of procedures are kept in the areas designated for use in response to an emergency. These are not complete sets, but contain only those procedures that are used to implement activities in the location where they are kept. Please dispose of any sections distributed that are not tabbed in the indicated copy.

C. Long - RAF/RPO (106, 107)
C. Long - SBF/ENV (108, 109)
C. Long - SBF/EM Team (110, 111, 111A)
C. Long – Aurora Medical Center (118, 119)
W. Flint - Cold Chem/HR Sample Room (113)
N. Deda - SBF/SEC (114)
M. Anderson - CR/Communicator (116)(Partial Distribution)
Simulator/Communicator (117)
J. Fletcher - Security (121)
N. Deda - Security Building (120)
K. Evers (125)
J. Stoeger (126)

Originals to KNPP QA Vault

Please follow the directions when updating your EPIP Manual. WATCH FOR DELETIONS!!! These are controlled procedures and random checks may be made to ensure the manuals are kept up-to-date.

*THIS IS NOT A CONTROLLED COPY. *IT IS A COPY FOR INFORMATION ONLY.*

KEWAUNEE NUCLEAR POWER PLANT
REVISION OF EMERGENCY PLAN IMPLEMENTING PROCEDURES
July 18, 2000

Please follow the directions listed below. If you have any questions regarding changes made to the EPIPs, please contact Dave Seebart at ext. 8719. If you are a controlled copy holder (see cover page), return this page to Diane Fencl by August 18, 2000, SIGNED AND DATED to serve as a record of revision.

EPIP Index, dated 07-18-2000.

DELETE		INSERT	
PROCEDURE	REV.	PROCEDURE	REV.
EP-RET-5	F	EPIP-RET-05	G
EP-TSC-2	Q	EPIP-TSC-02	R
EP-TSC-10	G	EPIP-TSC-10	H

I CERTIFY Copy No._______ (WPSC No.) of the Kewaunee Nuclear Power Plant's EPIPs has been updated.

____
SIGNATURE **DATE**

Please return this sheet to *DIANE FENCL*.

Diane Fencl
Diane Fencl

Enclosure

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EPIP-AD-03	KNPP Response to an Unusual Event	AA	04-18-2000
EPIP-AD-04	KNPP Response to Alert or Higher	AB	04-18-2000
EP-AD-5	Site Emergency	**Deleted**	04-27-87
EP-AD-5	Emergency Response Organization Shift Relief Guideline	A	10-13-98
EP-AD-6	General Emergency	**Deleted**	04-24-87
EPIP-AD-07	Initial Emergency Notifications	AK	02-01-2000
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1.0 Purpose

1.1 This procedure provides instruction for determining the activity release rate from the steam reliefs and the resultant instantaneous dose rates at the site boundary during controlled cooldown of the plant assuming primary-to-secondary leakage in the steam generators and steam release to the environment.

2.0 General Notes

2.1 This procedure is to be used in conjunction with Integrated Plant Emergency Operating Procedure ECA-3.1.

3.0 Precautions and Limitations

3.1 Uncontrolled steam releases from the steam generators and applicable plume projection procedures can be found in EPIP-RET-02B and EPIP-ENV-03C.

3.2 Westinghouse guidance specifies that since the steam releases are controlled, the site boundary dose rate limits of the Offsite Dose Calculation Manual (ODCM) are to be followed rather than the emergency limits of Federal Regulation 10CFR100.

3.3 The instantaneous dose rate equations in Section 5.0 of this procedure relate to Equations 2.4 and 2.5 of the ODCM, where:

3.3.1 The default X/Q dispersion factor equals 3.6E-6.

3.3.2 Summation factors are the summation of the Total Body Dose Factors (Ki) and Skin Dose Factors (Li), listed in Table 2.1 of the ODCM, times the default X/Q. Isotopes Ar-41, Kr-90, and Xe-137 are not included since they do not appear in EPIP-RET-02B, Table 2B.1, "Isotope Normalization Factor."

3.4 The steam line monitor calibration factors are taken from EPIP-RET-02B which is based on Fluor's evaluation performed under DCR 844, Task #965 (KPS-6266) of 3/16/81.

4.0 Initial Conditions

4.1 This procedure shall be implemented upon declaration of an Alert, Site Emergency, General Emergency, or when directed by the Shift Supervisor or Emergency Director.

5.0 Procedure

5.1 Obtain the following information from the TSC or Control Room:

5.1.1 Reactor Trip time.

5.1.2 Steam release flow rate (cc/sec) from EPIP-TSC-08A, "Calculations for Steam Release from Steam Generators."

5.1.3 Steam line monitor readings from:

- R-31 1A Steam Line - LO (mR/hr)
- R-32 1A Steam Line - HI (R/hr)
- R-33 1B Steam Line - LO (mR/hr)
- R-34 1B Steam Line - HI (R/hr)

5.2 Using the following table, determine the correct steam line monitor calibration factor based on time since Reactor Trip:

TIME SINCE REACTOR TRIP	STEAM LINE MONITOR CALIBRATION FACTOR $\frac{\mu Ci/cc}{R/hr}$
0 hours	14.5
1 hour	16.7
2 hours	20.3
4 hours	30.4
8 hours	67.9
1 day	887.0
1 week	3.08E + 4
1 month	1.93E + 4

5.3 Calculate the activity release rate as follows:

HIGHEST MONITOR READING		STEAM LINE MONITOR CAL. FACTOR		STEAM RELEASE FLOW RATE		1.06E-6	=	Ci/sec
R/hr		$\frac{\mu Ci/cc}{R/hr}$		cc/sec		Ci/μCi		
___________	x	___________	x	___________	x	1.0E-6	=	___________

5.4 Insert the activity release rate (Ci/sec) into the following equation to calculate the instantaneous total body dose rate at the site boundary:

ACTIVITY RELEASE RATE	x	SUMMATION FACTOR	=	INSTANTANEOUS DOSE RATE TO THE TOTAL BODY IN MREM/YEAR
(Ci/sec)				
___________________	x	0.190	=	______________ mRem/yr (TB)

5.5 The ODCM 3.4.1.a limit is 500 mRem/yr for the total body instantaneous dose rate.

5.6 Determine the dose rate to the skin as follows:

ACTIVITY RELEASE RATE	x	SUMMATION FACTOR	=	INSTANTANEOUS DOSE RATE TO SKIN IN MREM/YEAR
(Ci/sec)				
___________________	x	0.339	=	___________ mRem/yr (SKIN)

5.7 The ODCM 3.4.1.a limit is 3,000 mRem/yr for the instantaneous dose rate to the skin.

6.0 Final Conditions

6.1 Plant Emergency has been Terminated or Recovery actions have begun and the Emergency Response Manager has suspended the use of EPIPs.

7.0 References

7.1 KNPP Offsite Dose Calculation Manual

7.2 EPIP-RET-02B, Gaseous Effluent Sample and Analysis

7.3 IPEOP ECA-3.1, SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

7.4 EPIP-TSC-08A, Calculations for Steam Release from Steam Generators

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

- EPIP-RET-05 Procedures, Completed

8.1.2 Non-QA Records

None

1.0 Purpose

1.1 This procedure provides instruction for ensuring that the required actions are taken to enable the Technical Support Center (TSC) to provide support in a declared emergency.

2.0 General Notes

2.1 Functional support provided by the TSC need not be delayed waiting for full activation of the TSC.

3.0 Precautions and Limitations

3.1 None

4.0 Initial Conditions

4.1 The Technical Support Center (TSC) is activated for an Alert, Site Emergency, General Emergency, or at the request of the Emergency Director.

5.0 Procedure

5.1 Technical Support Center Director (TSCD) OR until a TSCD arrives, an appropriate TSC staff member shall:

5.1.1 Obtain the TSC Activation Manual from the TSC materials/supplies cabinet.

5.1.2 Direct available TSC staff members to initiate each of the following checklists:

Note
Initiation of the ERDS link must be completed within ***ONE HOUR*** *after declaration of an Alert or higher classification*

Note
NRC Communicators are most proficient at performing the ERDS Activation Checklist.

Note

Once the ERDS has been initiated, no further attention to the system should be required. Requests to restart or reset the system from the NRC must be filled promptly.

a. ______ TSC and OSF Activation Checklist, Form EPIPF-TSC-02-01

b. ______ TSC Ventilation Checklist, Form EPIPF-TSC-02-02

c. ______ Emergency Response Data System (ERDS) Link Initiation Checklist, Form EPIPF-TSC-02-03

d. ______ TSC Chart Recorder Operation Checklist, Form EPIPF-TSC-02-04

5.1.3 Ensure the following TSC positions are staffed (Refer to EPIP-APPX-A-2).

Note

As positions are filled, they should be signed in on the appropriate status board and be wearing their arm bands.

a. ______ Technical Support Center Director (TSCD)

b. ______ Radiological Protection Director (RPD)

c. ______ Support Activities Director (SAD)

d. ______ Site Protection Director (SPD)

e. ______ Engineering Coordinator (ENGCd)

f. ______ Quality Control Coordinator QCCd)

g. ______ Data Coordinator (DATACd)

h. ______ SAM Team Leader (SAMTL)

i. ______ SAM Core Hydraulics (SAMCH)

j. ______ SAM Operations (SAMOps)

k. ______ NRC Communicator (NRCCm)

l. ______ Operations Communicator (OpsCm)

m. ______ EOF Communicator (EOFCm)

n. ______ Accountability Area Coordinator (ACCCd)

o. ______ TSC Support Person (TSC-S)

5.1.4 As the TSC is capable of assuming responsibility for each of the following functions, inform the Control Room and/or EOF.

Note
IF the EOF Communicator is responsible for off-site notifications, THEN verify ED is in the TSC and concurs with this status.

a. ______ Off-site notifications (if EOF is not activated)

b. ______ NRC communications (ENS/ERDS)

Note
The Communicator conference call should be functioning to consider this position functional.

c. ______ EOF communication

d. ______ EAL assessment

e. ______ Accident assessment

5.1.5 Confirm and verify radiological habitability in the TSC with the Radiological Protection Director (RPD).

5.1.6 Periodically announce TSC functional and activation status to the TSC/OSF/RAF **AND** report the same to the Control Room and EOF.

5.1.7 WHEN the TSC is capable of performing all the functions defined in step 5.1.2, inform the Control Room and EOF that the TSC is fully activated.

5.1.8 WHEN the Emergency Director has determined that the declared emergency can be closed out and the TSC as a facility is no longer needed to support Control Room or off-site activities, secure the TSC using "TSC and OSF De-Activation Checklist," Form EPIPF-TSC-02-05.

6.0 Final Conditions

6.1 Plant Emergency has been closed out AND the TSC is no longer needed for plant or off-site recovery activities.

7.0 References

7.1 COMTRAK 93-025, Item 1

7.2 EPIP-AD-03, KNPP Response to an Unusual Event

7.3 EPIP-AD-04, KNPP Response to Alert or Higher

7.4 EPIP Appendix A, Communications

7.5 EPIP Appendix B, Forms EPIPF-TSC-02-01, TSC-02-02, TSC-02-03, TSC-02-04, and TSC-02-05

7.6 Kewaunee Nuclear Power Plant Emergency Plan

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 <u>QA Records</u>

- TSC and OSF Activation Checklist, Form EPIPF-TSC-02-01
- TSC Ventilation Checklist, Form EPIPF-TSC-02-02
- Emergency Response Data System (ERDS) Link Initiation Checklist, Form EPIPF-TSC-02-03
- TSC Chart Recorder Operation Checklist, Form EPIPF-TSC-02-04
- TSC and OSF De-Activation Checklist, Form EPIPF-TSC-02-05

8.1.2 <u>Non-QA Records</u>

None

1.0 Purpose

1.1 This procedure provides instruction for technical guidance to Severe Accident Management Operations (SAMOPs) and Technical Support Center staff members during implementation of the Kewaunee Nuclear Power Plant (KNPP) Integrated Plant Emergency Operating Procedures (IPEOPs).

2.0 General Notes

2.1 This procedure applies to steps in the IPEOPs that require consultation with the Plant Technical Support Engineering staff.

3.0 Precautions and Limitations

3.1 This procedure should only be used as a guide to the Technical Support staff. Plant parameters should be monitored to determine plant conditions prior to implementation of these guidelines.

4.0 Initial Conditions

4.1 This procedure is used during a declared emergency when plant conditions require assistance by the Technical Support staff in the execution of IPEOPs.

5.0 Procedure

5.1 IPEOP E-1: LOSS OF REACTOR OR SECONDARY COOLANT

5.1.1 Evaluate Plant Status:

a. This step instructs the operator to consult Technical Support staff to determine if E-MDS-30, Post-Accident Leakage Control, should be implemented. Post-Accident Leakage Control is actuated if Auxiliary Building radiation levels are increasing or significant core damage has occurred. Determination of whether E-MDS-30 is to be implemented should be made at this time because the procedure requires local actions which may be prohibited following transfer to Containment sump recirculation due to high radiation levels.

b. Chemistry is contacted to start up the Containment Hydrogen Monitoring System (EPIP-RET-03C) and obtain primary and secondary samples per other EPIPs.

5.2 IPEOP E-1: LOSS OF REACTOR OR SECONDARY COOLANT

5.2.1 Determine if Reactor Vessel Head Should Be Vented:

a. The possibility exists for a noncondensable bubble to form in the reactor vessel head region during certain LOCA events (whenever saturation conditions exist in the vessel head or gas is injected into or generated within the RCS). The reactor vessel head might have to be vented using the Reactor Vessel Head Vent System to prevent the bubble from growing to the extent that core cooling flow is adversely affected.

5.2.2 Identify Growth of a Void in the Vessel:

a. The growth of a void in the vessel upper head can be identified by monitoring the Reactor Vessel Liquid Inventory System (RVLIS) upper range. A RVLIS indicating less than a full upper head is the primary means of determining if voids exist. In addition to RVLIS, other indirect indications of voids in the RCS are listed below (these voids are not necessarily located in the reactor vessel head).

1. Pressurizer level response to RCS pressure changes may not be normal if voids exist in the RCS. The pressurizer level may decrease during a RCS pressurization due to void compression or condensation. Also, the level may rise rapidly during a spraying operation due to void expansion or generation.

2. An indication of reactor vessel head temperatures equal to or greater than saturation temperature warrants the assumption that a steam bubble has been generated in the reactor vessel head.

3. The operator may suspect noncondensable voids in the RCS after either a complete SI accumulator tank discharge or an inadequate core cooling condition.

b. <u>IF</u> a steam void is formed during post-LOCA cooldown and depressurization or during a steam generator tube rupture recovery, <u>THEN</u> no attempt should be made to condense the void through repressurization. Only RXCP restart or continued cooling from CRDM fans should be used. Refer to IPEOP FR-I.3 and Section 5.26 of this procedure.

5.3 IPEOP E-1: LOSS OF REACTOR OR SECONDARY COOLANT

5.3.1 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.4 IPEOP E-1: LOSS OF REACTOR OR SECONDARY COOLANT

5.4.1 Evaluate Long-Term Plant Status:

a. The equipment needed to function following an event has been designed so that operation for extremely long periods of time is possible. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

b. Actions must be taken to adjust recirculation sump pH to between 8 and 10.5 within 48 hours of the start of the leak to prevent component stress corrosion cracking.

5.5 IPEOP ES-1.2: POST-LOCA COOLDOWN AND DEPRESSURIZATION

5.5.1 Check if RHR System Should Be Placed in Service:

a. The RHR System is designed to operate below specific RCS pressure and temperature conditions (RCS hot leg temperature less than 400°F and RCS pressure less than 425 psig). Depending on the size of the break, different actions should be taken.

b. For smaller breaks, the SI pumps will have been stopped in most cases and most of the RWST water will still be available by the time the RHR System entry criteria are satisfied. For these cases, the RHR System could be placed in service with the RHR pumps taking suction from the hot legs. Any high-head pump left running would remain aligned in the cold leg injection mode taking suction from the RWST. When charging flow is established, the injection source is also from the RWST.

c. For larger breaks, the RWST level will eventually decrease to the recirculation transfer setpoint and at least one RHR pump must be used for containment sump recirculation. IF the RHR System is not placed in service, THEN the system can remain in the long-term recirculation mode with the core residual heat being dissipated through the safeguards (RHR) heat exchangers.

5.5.2 Consider These Three Important Factors:

a. The RWST (or alternate) source of injection (make-up) water must be available for operating high-head SI, charging pumps, and RHR in split-train operation.

b. Confirmation of system availability, including all pumps, valves, and adequate inventory in the RCS to preclude steam from entering the RHR pump suction, must take place before RHR operation can begin.

c. Auxiliary building radiation levels should be evaluated. Placing RHR in service in the normal lineup will cause potentially highly radioactive fluid to be transported through lines that did not have radioactive fluid in them prior to the event. Care should be taken to minimize the spread of radioactive fluid through the CVCS system, if possible. Additionally, during a design basis LOCA, some valves and equipment (such as RHR-10A and RHR-10B) are projected to be in radiation fields of 1,000 R/hr or more due to "shine" from the containment building.

5.6 IPEOP ES-1.2: POST-LOCA COOLDOWN AND DEPRESSURIZATION

5.6.1 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.7 IPEOP ES-1.2: POST-LOCA COOLDOWN AND DEPRESSURIZATION

5.7.1 Evaluate Long-Term Plant Status:

a. After reaching and maintaining cold shutdown conditions, the plant is effectively stable for the long term. IF the SI pumps were stopped, THEN RCS subcooling would have been restored and RCS circulation flow should have been adequate to prevent boron precipitation. Thus, the transfer of hot leg recirculation would probably not be needed for the smaller breaks where SI flow was reduced.

5.8 IPEOP ES-3.1: POST-SGTR COOLDOWN USING BACKFILL

5.8.1 Evaluate Long-Term Plant Status:

a. The equipment needed to function following an event has been designed so that operation for extremely long periods of time is possible. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.9 IPEOP ES-3.2: POST-SGTR COOLDOWN USING BLOWDOWN

5.9.1 Evaluate Long-Term Plant Status:

a. The equipment needed to function following an event has been designed so that operation for extremely long periods of time is possible. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.10 IPEOP ES-3.3: POST-SGTR COOLDOWN USING STEAM DUMP

5.10.1 Evaluate Long-Term Plant Status:

a. The equipment needed to function following an event has been designed so that operation for extremely long periods of time is possible. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.11 IPEOP ECA-0.0: LOSS OF ALL AC POWER

5.11.1 IF core exit temperatures are greater than 1200°F and increasing, THEN go to SACRG-1, Severe Accident Control Room Guideline Initial Response.

a. The Severe Accident Management Guidelines (SAMGs) are entered from the ERGs by Control Room Operators when core damage occurs. The ERG to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the ERGs to the SAMGs. The 1200°F criteria for transition from the ERGs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

b. IF the Operator enters this step and core exit TC temperatures are greater than 1200°F and increasing, THEN the Operator should transition to the SAMGs. This condition indicates that all attempts to restore core cooling have failed, core damage cannot be prevented, and the Operator should go to the SAMGs.

5.12 IPEOP ECA-1.1: LOSS OF EMERGENCY COOLANT RECIRCULATION

5.12.1 Consult with Emergency Director to determine if RHR System should be placed in service.

a. The RHR System is designed to operate below specific RCS pressure and temperature conditions. IF previous actions to establish conditions were not complete, THEN this step directs the Operator to continue with the procedure for completion of the actions. At this time, the plant staff should determine RHR System availability. RHR System availability includes confirmation of equipment needed for RHR System operation (RHR suction valves, RHR pumps, etc.) and confirmation of adequate liquid inventory in the RCS to preclude steam from entering the RHR pump suction.

5.12.2 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.12.3 Consult with Emergency Director:

a. This procedure provides generic instructions for cooldown and depressurization of the plant to atmospheric conditions following a loss of emergency coolant recirculation. After the steps have been completed and cold shutdown conditions have been maintained, the Plant Engineering staff has time to evaluate the event and develop recovery procedures so that the Plant can be repaired and brought back to service.

5.13 IPEOP ECA-2.1: UNCONTROLLED DEPRESSURIZATION OF BOTH SGs

5.13.1 Evaluate Long-Term Plant Status:

a. The equipment needed to function following an event has been designed so that operation for extremely long periods of time is possible. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.14 IPEOP ECA-3.1: SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

5.14.1 Consult with Emergency Director:

a. This step instructs the Operator to consult with the Emergency Director when ruptured SG narrow range level exceeds 92%. An inability to prevent SG overfill may result from a rupture large enough to require the use of ECA-3.2, "SGTR with Loss of Reactor Coolant - Saturated Recovery Desired."

5.15 IPEOP ECA-3.1: SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

5.15.1 Check if RHR System Should Be Placed in Service:

a. The RHR System is designed to operate below specific RCS pressure and temperature conditions (RCS hot leg temperature less than 400°F and RCS pressure less than 425 psig). When such conditions are established, the RHR System should be placed in service to complete the cooldown to cold shutdown and provide long-term cooling.

5.15.2 Consider These Three Important Factors:

a. The RWST (or alternate) source of injection (makeup) water must be available for operating high-head SI, charging pumps, and RHR in split-train operation.

b. Confirmation of system availability including all pumps, valves, and adequate inventory in the RCS to preclude steam from entering the RHR pump suction must take place before RHR operation can begin.

c. Auxiliary building radiation levels should be evaluated. Placing RHR in service in the normal lineup will cause potentially highly radioactive fluid to be transported through lines that did not have radioactive fluid in them prior to the event. Care should be taken to minimize the spread of radioactive fluid through the CVCS System if possible. Additionally, during some design basis accidents, some valves and equipment (such as RHR-10A and RHR-10B) are projected to be in radiation fields of 1,000 R/hr or more due to "shine" from the containment building.

5.16 IPEOP ECA-3.1: SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

5.16.1 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.17 IPEOP ECA-3.1: SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

5.17.1 Evaluate Long-Term Plant Status:

a. After reaching and maintaining cold shutdown conditions, the plant is effectively stable for the long term. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.18 IPEOP ECA-3.2: SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED

5.18.1 Check if RHR System Should Be Placed in Service:

a. The RHR System is designed to operate below specific RCS pressure and temperature conditions (RCS hot leg temperature less than 400°F and RCS pressure less than 425 psig). When such conditions are established, the RHR System should be placed in service to complete the cooldown to cold shutdown and provide long-term cooling.

5.18.2 Consider These Three Important Factors:

a. The RWST (or alternate) source of injection (make-up) water must be available or operating high-head SI, charging pumps, and RHR in split-train operation.

b. Confirmation of system availability including all pumps, valves, and adequate inventory in the RCS to preclude steam from entering the RHR pump suction must take place before RHR operation can begin.

c. Auxiliary building radiation levels should be evaluated. Placing RHR in service in the normal lineup will cause potentially highly radioactive fluid to be transported through lines that did not have radioactive fluid in them prior to the event. Care should be taken to minimize the spread of radioactive fluid through the CVCS System if possible. Additionally, during some design basis accidents, some valves and equipment (such as RHR-10A and RHR-10B) are projected to be in radiation fields of 1,000 R/hour or more due to "shine" from the containment building.

5.19 IPEOP ECA-3.2: SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED

5.19.1 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.20 IPEOP ECA-3.2: SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED

5.20.1 Evaluate Long-Term Plant Status:

a. After reaching and maintaining cold shutdown conditions, the plant is effectively stable for the long term. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.21 IPEOP ECA-3.3: SGTR WITHOUT PRESSURIZER PRESSURE CONTROL

5.21.1 Evaluate Long-Term Plant Status:

a. After reaching and maintaining cold shutdown conditions, the plant is effectively stable for the long term. This allows the Plant Engineering staff time to evaluate the event and develop recovery procedures so that the plant can be repaired and brought back to service. Priority should be given, however, to ensure that equipment needed for accident mitigation remains operable.

5.22 IPEOP FR-S.1: RESPONSE TO NUCLEAR POWER GENERATION/ATWS

5.22.1 IF core exit temperatures are greater than 1200°F and increasing, THEN go to SACRG-1, Severe Accident Control Room Guideline Initial Response.

a. The Severe Accident Management Guidelines (SAMGs) are entered from the ERGs by Control Room Operators when core damage occurs. The ERG to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the ERGs to the SAMGs. The 1200°F criteria for transition from the ERGs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

b. IF the Operator enters this step and core exit TC temperatures are greater than 1200°F and increasing, THEN the Operator should transition to the SAMGs. This condition indicates that all attempts to restore core cooling have failed, core damage cannot be prevented, and the Operator should go to the SAMGs.

5.23 IPEOP FR-C.1: RESPONSE TO INADEQUATE CORE COOLING

5.23.1 IF core exit TC temperatures increasing AND RXCPs running in all available RCS cooling loops, THEN go to SACRG-1, Severe Accident Control Room Guideline Initial Response.

a. The Severe Accident Management Guidelines (SAMGs) are entered from the ERGs by Control Room Operators when core damage occurs. The ERG to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the ERGs to the SAMGs. The 1200°F criteria for transition from the ERGs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

b. IF the Operator enters this step and core exit TC temperatures are greater than 1200°F and increasing and all available RXCPs are running, THEN the Operator should transition to the SAMGs. This condition indicates that all attempts to restore core cooling have failed, core damage cannot be prevented, and the Operator should go to the SAMGs.

5.23.2 Consult with Emergency Director for Additional Recovery Actions:

a. This step instructs the operator to notify the Emergency Director when the hydrogen concentration inside containment is greater than 6% in air. The possible actions to be taken with high hydrogen concentrations in containment are dependent on the containment conditions, the event progression, and off-site conditions.

b. Evaluate actions to be taken for high containment hydrogen concentration using SAG-7.

5.24 IPEOP FR-Z.2: RESPONSE TO CONTAINMENT FLOODING

5.24.1 Notify Emergency Director of Sump Level and Activity Level to Obtain Recommended Action:

a. The ED should request evaluation of the cause of the event and provide specific recommendations to the Operators for reducing containment water level.

5.24.2 Consider the Following Three Methods to Reduce Flooding:

a. Location of critical plant components in relation to containment sump water level.

b. Location, size, and shielding of available storage tanks outside containment.

c. Radiation concerns due to pump and line routing from the containment sump to the various storage tanks.

5.25 IPEOP FR-Z.3: RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL

5.25.1 Notify Emergency Director of Containment Radiation Level to Obtain Recommended Action:

a. After containment vent isolation has been verified, check the pressurizer water level, charging flow, and operation of the containment sump pumps to determine if a reactor coolant leak is occurring. IF there is a lack of evidence of a reactor coolant leak, THEN verify the alarm condition by selecting the fast advance on the air particulate and sample fresh air for about 15 seconds to confirm that the detector function is normal. IF it is normal, THEN notify the RPD.

b. An additional area to be looked at is the possibility of fuel damage. By checking the thermocouple readings, hydrogen generation level, and RCS activity levels, it can be determined whether or not damage to the fuel has occurred.

5.26 IPEOP FR-I.3: RESPONSE TO VOIDS IN REACTOR VESSEL

5.26.1 Obtain Maximum Allowable Venting Time from Technical Support Center Director (Per EPIP-TSC-07):

a. Calculation of the maximum allowable venting time is based on maintaining containment hydrogen concentration below 3% in dry air. The lower the initial hydrogen concentration, the longer the venting can continue. Procedure EPIP-TSC-07 describes the method of determining RCS venting time.

6.0 Final Conditions

6.1 This procedure may be terminated when the emergency has been closed out or recovery operations have been entered, the plant is stable, and Operations has determined that technical support of IPEOPs is no longer required.

7.0 References

7.1 Kewaunee Nuclear Power Plant Integrated Plant Emergency Operating Procedures

7.2 Westinghouse Owners Group Emergency Response Guidelines

7.3 SAG-7

8.0 Records

8.1 The following QA records and non-QA records are identified in this directive/procedure and are listed on the KNPP Records Retention Schedule. These records shall be maintained according to the KNPP Records Management Program.

8.1.1 QA Records

None

8.1.2 Non-QA Records

None