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Regulatory Docket File

April 11, 1975

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Mr. D. L. Ziemann, Chief
Operating Reactors - Branch 2
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Units 2 and 3
Quad-Cities Station Units 1 and 2
Proposed Amendment to Facility
Operating Licenses DPR-19, DPR-25,
DPR-29, and DPR-30
NRC Dkts. 50-237, 50-249,
50-254, and 50-265

Dear Mr. Ziemann:

In response to your letter dated February 14, 1975, attached are proposed amendments to facility operating licenses DPR-19, 25, 29, and 30, Appendix A, Technical Specifications. The purpose of the proposed amendments is to provide limiting conditions for operation which preclude torus structural damage due to "steam quenching vibration phenomena".

The proposed amendments are indicated on the attached revised Technical Specification pages:

DPR-19 and DPR-25	DPR-29 and DPR-30
108, 108a, 108b, 125, 125a, 129, and 129a	143, 143a, 143b, 144, 165, 165a, 169, and 169a

The temperature limits proposed are justified by the evaluation in reference (5) of your letter dated February 19, 1975.

These proposed amendments have received Onsite and Offsite review and approval.

Three (3) signed originals and 77 copies of this transmittal letter are submitted, and 40 copies of the proposed Technical Specification page changes are submitted for DPR-19 and 25, and DPR-29 and 30.

Very truly yours,

Abel for
R. L. Bolger
Assistant Vice-President

SUBSCRIBED and SWORN to
before me this 11th day
of April, 1975.

Nancy M. Hollingsworth
Notary Public

4440

Regulatory Docket File

DRESDEN STATION UNIT 2 4-11-75

DPR-19

Proposed Amendment

to

Technical Specification

Pages 108, 108a, 108b, 125, 125a, 129, and 129a

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The temperature and volume of the water in the suppression chamber shall at all times, except as specified in Specifications 3.5.F.3 or 3.5.F.4, be maintained within the following limits:

   a. Minimum water volume 112,000 ft^3

   b. Maximum water volume 115,655 ft^3

   c. Maximum water temperature

      1. During normal power operation: 95°F. If this limit is exceeded for any reason, the suppression pool temperature

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day. Whenever there is an indication that a significant amount of heat is being added to the suppression pool, the pool temperature shall be continually observed and recorded or logged every five minutes until heat addition is terminated. Whenever there is an indication of relief valve operation at reactor pressure above 150 psig and suppression pool temperatures above 160°F an external visual examination

3.7 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS

shall be reduced to 95°F or less within 48 hours or the reactor shall be placed in the cold shutdown condition.

2. During testing which adds heat to the suppression pool: 105°F. If this limit is exceeded for any reason, the tests shall be immediately terminated.

3. During reactor power operation: 110°F.

   (a) if this limit is exceeded and there is an indication of a stuck open relief valve, the reactor shall be manual scrammed.

   (b) if this limit is exceeded for any other reason, an orderly shutdown shall be initiated.

4. Following a scram or shutdown with the reactor pressurized above 150 psig: 120°F. If this

of the suppression chamber shall be conducted before power operation is resumed. The interior painted surfaces above the water line of the suppression pool shall be inspected at each refueling outage.

3.7 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
limit is exceeded, the reactor shall be depressurized at normal cool down rates or as necessary to reduce reactor pressure to 150 psig before suppression chamber temperatures reach 160°F.	
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).	2. The primary containment integrity shall be demonstrated by either Method A or Method B, as follows: a. Integrated Primary Containment Leak Test (IPCLT)

Bases:

3.7

A. Primary Containment — The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.3%Δk. A drop of a 1.3%Δk rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Ref. Section 5.2.3 SAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (9) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge

(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

3.7

exit. Specifications have been conservatively placed on the envelope of reactor operating conditions so that heat addition may be terminated or the reactor may be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. The need to relax these conservative limits to more realistic values is presently being evaluated.

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and

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(10) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

Bases:

4.7

A. Primary Containment

Due to the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly. Daily monitoring of these parameters is sufficient under normal conditions to establish undesirable trends. During periods of significant heat addition to the suppression pool, continuous observation and frequent logging of pool temperature ensures that temperature trends will be followed closely enough to permit appropriate action. The requirement for an external visual examination following an event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinities of relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay (12).

The design pressure of the drywell and absorption chamber is 62 psig (12). The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 2.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the

(12) Section 5.2 of the SAR.

Bases: (cont'd)

4.7

maximum total whole body passing cloud dose is about 8 rem and the maximum total thyroid dose is about 185 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 5 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

DRESDEN STATION UNIT 3

Received w/Ltr Dated 4-11-75

DPR-25

Proposed Amendment

to

Technical Specification

Pages 108, 108a, 108b, 125, 125a, 129, and 129a

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The temperature and volume of the water in the suppression chamber shall at all times, except as specified in Specifications 3.5.F.3 or 3.5.F.4, be maintained within the following limits:

   a. Minimum water volume 112,000 ft^3

   b. Maximum water volume 115,655 ft^3

   c. Maximum water temperature

      1. During normal power operation: 95°F. If this limit is exceeded for any reason, the suppression pool temperature

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day. Whenever there is an indication that a significant amount of heat is being added to the suppression pool, the pool temperature shall be continually observed and recorded or logged every five minutes until heat addition is terminated. Whenever there is an indication of relief valve operation at reactor pressure above 150 psig and suppression pool temperatures above 160°F an external visual examination

shall be reduced to 95°F or less within 48 hours or the reactor shall be placed in the cold shutdown condition.

2. During testing which adds heat to the suppression pool: 105°F. If this limit is exceeded for any reason, the tests shall be immediately terminated.

3. During reactor power operation: 110°F.

   (a) if this limit is exceeded and there is an indication of a stuck open relief valve, the reactor shall be manual scrammed.

   (b) if this limit is exceeded for any other reason, an orderly shutdown shall be initiated.

4. Following a scram or shutdown without initiating reactor depressurization 120°F. If this

of the suppression chamber shall be conducted before power operation is resumed. The interior painted surfaces above the water line of the suppression pool shall be inspected at each refueling outage.

3.7 LIMITING CONDITION FOR OPERATION

limit is exceeded, the reactor shall be depressurized at normal cool down rates or as necessary to reduce reactor pressure to 150 psig before suppression chamber temperatures reach 160°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated by either Method A or Method B, as follows:

   a. Integrated Primary Containment Leak Test (IPCLT)

Bases:

3.7

A. Primary Containment — The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.3%Δk. A drop of a 1.3%Δk rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Ref. Section 5.2.3 SAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet and the minimum volume of 112,000 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (9) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge

(9) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases: (cond't

3.7

exit. Specifications have been conservatively placed on the envelope of reactor operating conditions so that heat addition may be terminated or the reactor may be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. The need to relax these conservative limits to more realistic values is presently being evaluated.

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and

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(10) Robbins, C. H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

Bases:

4.7

A. Primary Containment

Due to the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly. Daily monitoring of these parameters is sufficient under normal conditions to establish undesirable trends. During periods of significant heat addition to the suppression pool, continuous observation and frequent logging of pool temperature ensures that temperature trends will be followed closely enough to permit appropriate action. The requirement for an external visual examination following an event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinities of relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay (12).

The design pressure of the drywell and absorption chamber is 62 psig (12). The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 2.0%/day at 48 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the

(12) Section 5.2 of the SAR.

Bases: (cont'd)

4.7

maximum total whole body passing cloud dose is about 8 rem and the maximum total thyroid dose is about 185 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 5 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
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April 11, 1975

Regulatory Docket File

Mr. D. L. Ziemann, Chief
Operating Reactors - Branch 2
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

4-11-75

Subject: Dresden Station Units 2 and 3
Quad-Cities Station Units 1 and 2
Proposed Amendment to Facility
Operating Licenses DPR-19, DPR-25,
DPR-29, and DPR-30
NRC Dkts. 50-237, 50-249,
50-254, and 50-265

Dear Mr. Ziemann:

In response to your letter dated February 14, 1975, attached are proposed amendments to facility operating licenses DPR-19, 25, 29, and 30, Appendix A, Technical Specifications. The purpose of the proposed amendments is to provide limiting conditions for operation which preclude torus structural damage due to "steam quenching vibration phenomena".

The proposed amendments are indicated on the attached revised Technical Specification pages:

DPR-19 and DPR-25	DPR-29 and DPR-30
108, 108a, 108b, 125, 125a, 129, and 129a	143, 143a, 143b, 144, 165, 165a, 169, and 169a

The temperature limits proposed are justified by the evaluation in reference (5) of your letter dated February 19, 1975.

These proposed amendments have received Onsite and Offsite review and approval.

Three (3) signed originals and 77 copies of this transmittal letter are submitted, and 40 copies of the proposed Technical Specification page changes are submitted for DPR-19 and 25, and DPR-29 and 30.

Very truly yours,

R. L. Bolger for
R. L. Bolger
Assistant Vice-President

SUBSCRIBED and SWORN to
before me this 11th day
of April, 1975.

Nancy M. Hollingsworth
Notary Public

QUAD-CITIES STATION UNITS 1 AND 2

DPR-29 AND DPR-30

Proposed Amendment

to

Technical Specification

Pages 143, 143a, 143b, 144, 165, 165a, 169, and 169a.

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The temperature and volume of the water in the suppression chamber shall at all times, except as specified in Specifications 3.5.F.2 or 3.5.F.3, be maintained within the following limits:

   a. Minimum water volume 112,200 ft^3 shown on level indicator as -2.0 inches.

   b. Maximum water volume during normal power operation 115,655 ft^3 shown on level indicator as +2.0 inches.

   c. Maximum water temperature

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day. Whenever there is an indication that a significant amount of heat is being added to the suppression pool, the pool temperature shall be continually observed and recorded or logged every five minutes until heat addition is terminated. Whenever there is an indication of relief valve operation at reactor pressure above 150 psig and suppression pool temperatures above 160°F an external visual examination

3.7 LIMITING CONDITION FOR OPERATIONS	4.7 SURVEILLANCE REQUIREMENTS
1. During normal power operation: 95°F. If this limit is exceeded for any reason, the suppression pool temperature shall be reduced to 95°F or less within 48 hours or the reactor shall be placed in the cold shutdown condition. 2. During testing which adds heat to the suppression pool: 105°F. If this limit is exceeded for any reason, the tests shall be immediately terminated. 3. During reactor power operation: 110°F. (a) if this limit is exceeded and there is an indication of a stuck open relief valve, the reactor shall be manual scrammed. (b) if this limit is exceeded for any other reason, an orderly shutdown shall be initiated.	of the suppression chamber shall be conducted before power operation is resumed. The interior painted surfaces above the water line of the suppression pool shall be inspected at each refueling outage.

3.7 LIMITING CONDITION FOR OPERATIONS	4.7 SURVEILLANCE REQUIREMENTS
4. Following a scram or shutdown with the reactor pressurized above 150 psig: 120°F. If this limit is exceeded, the reactor shall be depressurized at normal cool down rates or as necessary to reduce reactor pressure to 150 psig before suppression chamber temperatures reach 160°F.	

3.7 LIMITING CONDITION FOR OPERATION

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2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MW(t).

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated by conducting Integrated Primary Containment Leak Tests (IPCLT).

   a. IPCLT shall be performed at an initial pressure of approximately 48 psig, Pt(48).

   b. If local leak rate measurements are made prior to IPCLT, and repairs are found to be necessary and retests conducted, the leak rate difference, prior to and after repair when corrected to Pt(48) shall be added to the final integrated leak rate result.

   c. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

Using the minimum of maximum water volume given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 56 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet and the minimum volume of 112,200 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests(9) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been conservatively placed on the envelope of reactor operating conditions so that heat addition may be terminated or the reactor may be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. The need to relax these conservative limits to more realistic values is presently being evaluated.

The maximum temperature at the end of blowdown tested during the Humboldt Bay(10) and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for emergency core cooling systems operability as explained in Basis for Specification 3.5.F.

Using a 50°F rise (SAR Section 5.2.3.1) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (2 RHR pumps and 2 RHR service

(9) "Bodega Bay Preliminary Hazards Summary Report," Appendix 1, Docket 50-205, December 28, 1962.

(10) Robbins, C.H., "Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

3.7 Limiting Conditions for Operation Bases (cont'd)

water pumps) containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI mode of the RHR, and HPCI pumps.

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 56 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

3.7 Limiting Conditions for Operation Bases (cont'd)

C. Secondary Containment - The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling, except, however, for initial fuel loading of Unit 1 prior to initial power testing. Ref. SAR Section 1.

D. Primary Containment Isolation Valves - Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

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(11) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, ORNL-4071."

4.7 Surveillance Requirements Bases

A. Primary Containment - Due to the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly. Daily monitoring of these parameters is sufficient under normal conditions to establish undesirable trends. During periods of significant heat addition to the suppression pool, continuous observation and frequent logging of pool temperature ensures that temperature trends will be followed closely enough to permit appropriate action. The requirement for an external visual examination following an event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinities of relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

3.7 Limiting Conditions for Operation Bases (cont'd)

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 48 psig which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay.[12]

The design pressure of the drywell and absorption chamber is 56 psig[12]. The design leak rate is 0.5%/day at a pressure of 56 psig. As pointed out above, the pressure response of the drywell

(12) Section 5.2 of the SAR.