14.3.5 Environmental Consequences Of A Loss-Of-Coolant Accident

See Unit 1 Section 14.3.5.

14.3.6 Deleted

14.3.7 Long Term Cooling

This section has been revised to incorporate updated analytical material (Reference 5) regarding pressurized thermal shock (PTS) and the prevention of Reactor Coolant System (RCS) overpressurization conditions during, or subsequent to, periods of rapid and/or prolonged system cooldown. This material indicates that certain small LOCAs and double-ended (or equivalent) SGTRs with high break flow rates have replaced large steamline breaks, certain small break LOCAs and feedwater line breaks as the dominant PTS related accidents. General discussions of the PTS issues, including stagnant loop concerns and Emergency Response Guideline (ERG) actions are presented. The methodology followed in applying this generic information specifically to the Donald C. Cook Nuclear Plant is summarized. The PTS screening criteria and limiting approved calculated PTS values projected to the end of vessel life determined in accordance with 10 CFR 50.61 are also provided for Donald C. Cook Units 1 and 2. Finally, a brief general discussion of long term cooling and the use of WOG-ERG based plant emergency operating procedures to prevent excessive cooldown and reduce any PTS related risk is presented.

1. *Generic Background and Description*

A combination of severe cooling (thermal shock) and high pressure produces the condition that is called pressurized thermal shock (PTS). Within the thick walls of the reactor pressure vessel, a substantial temperature gradient can be produced by rapid cooling of the inner surface. This gradient results in thermal stresses that are tensile in nature and that are a maximum at the inner surface of the vessel. If the system is pressurized during or after the cooldown occurs, an additional pressure stress is imposed on the vessel wall, again being tensile in nature and having a maximum at the inner surface. It is this combined pressure-temperature stress that is of primary concern for PTS.

A limiting PTS condition that may challenge the reactor vessel integrity can occur during a severe transient such as a loss of coolant accident (LOCA), a secondary side depressurization (steamline or feedline break), or a steam generator tube rupture (SGTR). Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside of the vessel wall followed by high repressurization,
- significant degradation of vessel material toughness caused by radiation embrittlement, and
- the presence of a critical-size defect in the vessel wall.

A PTS concern arises if one of these transients acts on the highly irradiated beltline region of a reactor vessel where a reduced fracture resistance exists because of the neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Westinghouse Owners Group (WOG) completed a generic evaluation program of the severity of the thermal shock transient and submitted a report to the NRC, WCAP-10019, dated December 1981 (Reference 1). This generic report concluded that, based on a conservative assessment, all of the Westinghouse Pressurized Water Reactors, including Donald C. Cook Units 1 and 2, could continue to operate for a considerable number of years before the reactor vessel integrity acceptance criteria would be violated. The results of the WOG program demonstrated that no immediate reactor vessel integrity concerns exist.

As part of this continuing effort another report was submitted to the NRC in May 1982 by the Westinghouse Owners Group (Reference 2). This report provided additional information on WCAP-10019 and responded to the NRC's short term action needs as the Staff perceived them (Reference 3). Several methodological differences exist between References (1) and (2), particularly on the subject of crack arresting as the basis on which to predicate vessel integrity.

In the above studies, and also a probabilistic transient evaluation by the NRC (Reference 4), it was recognized that transients leading to stagnation of flow in the reactor coolant loops while safety injection flow continues may be an additional candidate contributing to PTS. The stagnant loop considerations are discussed below.

2. General Discussion of PTS Risk Including Stagnant Reactor Coolant Loop Conditions

During a transient or emergency event, all reactor coolant pumps (RCPs) may be stopped due to loss of support conditions (e.g., offsite power supply, cooling water to motors or seals) or due to meeting the RCP trip criteria. In this latter situation, the operator would be directed to trip the RCPs if a safety injection pump is running and the RCP trip parameter is reached (e.g., low RCS subcooling or low RCS pressure). After RCP trip, unless the residual heat removal (RHR) system is in service and is removing decay heat, natural circulation flow will be needed to remove core decay heat through the steam generators.

If natural circulation flow decreases or is stopped in one or more loops and safety injection (SI) flow is maintained to the cold legs of the affected loops, the relatively cold SI water will mix with the water in the cold legs and vessel downcomer. This can cause a PTS concern for the reactor vessel if the RCS pressure remains or becomes high.

For a rigorous determination of the PTS risk for the plant, literally thousands of postulated cooldown scenarios could be considered. However, by applying appropriately conservative approximations, it was possible to focus on the limiting cases and also analyze this problem on a generic basis. A generic study of PTS risk, including stagnant loop considerations, was performed by the Westinghouse Owners Group and is provided in WCAP-10319 (Reference 5).

The PTS study including stagnant loop conditions consisted of three main efforts:

- an event tree analysis of the seven transient families believed to include all the potential stagnant loop transients that contribute to the overall PTS risk.
- a thermal hydraulic analysis to determine a characteristic pressure, final temperature (reflecting the depth of the cooldown), and time constant (reflecting the rate of cooldown) for each of the unique sequences or "bins" identified in the event tree analysis, and
- application of the NRC probabilistic fracture mechanics (PFM) model from Reference (4) to determine the PTS risk associated with each of the identified sequences or bins.

In the WCAP-10319 study, the PTS risk is defined based on the frequency of significant flaw extension for longitudinal flaws that may exist at the inner surface of the reactor vessel. As noted below, welds oriented in the circumferential direction have a less stringent PTS screening criterion (i.e., 300 versus 270°F), so selection of flaws oriented in the axial or longitudinal direction for a measure of the PTS risk is appropriate and conservative. This selected PTS risk parameter is evaluated as a function of the mean surface RT_{NDT} (reference temperature nil-ductility transition). This vessel surface RT_{NDT} parameter, now referred to as the vessel RT_{PTS} (reference temperature for pressurized thermal shock), is provided in Figure 14.3.7-1. This figure can be applied to most of the WOG member plants including the Donald C. Cook Nuclear Plant.

In Figure 14.3.7-1, the PTS contributions associated with each of the seven categories or transient types investigated in the WCAP-10319 WOG study are provided. The corresponding "WOG TOTAL" is also shown and compares closely with the "NRC TOTAL" from Reference (Reference 4). The results of this WOG study show that the overall PTS risk for a typical Westinghouse plant is dominated by LOCAs and SGTRs, specifically small LOCAs with equivalent diameters in a certain range (about 2" to 6" for a 4-loop plant) and double-ended (or equivalent) SGTRs that occur simultaneously with a loss of offsite power or other conditions resulting in a trip of all RCPs.

Hot leg LOCAs were specifically analyzed for the generic study to maximize the amount of cold safety injection and accumulator water added to the RCS cold legs and vessel downcomer regions. Note that for smaller LOCA cases, SI flow would be able to keep up with break flow; if the RCPs are tripped for these cases, natural circulation flow would be maintained and there would be no uncontrolled cooldown of the cold legs and vessel downcomer regions. For larger break sizes, the cooldown would be uncontrolled but the RCS depressurizes quickly. Thus, for these extremes in break sizes, either the temperature or pressure stress contributions are minimized, so these events become less severe from the PTS perspective than LOCAs in the 2" to 6" range analyzed for the PTS studies.

Some of the scenarios used for the SGTR cases also account for extended delays in termination of safety injection following the initiation of operator actions to cooldown and depressurize the RCS (Note: In addition to identifying and isolating the ruptured steam generator, these operator actions are credited for the SGTR event). Manual SI termination delays following depressurization of the RCS to the ruptured S/G pressure were assumed for these cases. This delay tends to maximize the time that flow in the ruptured loop is slowed down or stagnates prior to SI termination. The maximum delay cases correspond to a total transient time of more than 60 minutes for operator action to terminate SI flow. This category of SGTR is calculated to contribute less than 1% to the estimated total SGTR initiating event frequency of 3.9 x 10^{-2} occurrences per reactor-year.

It is also important to point out that in recent years, there has been an increased awareness of the need for performing any required operator SGTR accident response actions in a timely manner. This increased operator awareness, combined with a high probability that the RCPs would be left operating for a design basis SGTR event, tends to make the SGTR contribution to overall PTS risk in WCAP-10319 conservatively high when applied to most WOG plants including the Donald C. Cook Nuclear Plant.

The WOG study indicated that the cold legs and vessel downcomer do not, on the average, cool down as much for the SGTR cases as for small LOCAs. It is because of this that below an RT_{PTS} of 290°F, small LOCAs are the dominant contributors to PTS risk, despite the greater initiating event frequency of the SGTR cases. At RT_{PTS} values above 290°F, SGTRs become the dominant contributor to PTS risk. A similar trend could be expected for the Donald C. Cook Nuclear Plant since the initiating event frequencies used in the generic PTS studies are comparable to or bounding when compared to those expected at the Cook plant.

Besides small LOCA and SGTR, the next most limiting event identified in Figure 14.3.7-1 is the loss of heat sink transient. This event is actually treated as a small hot leg LOCA since the operator would, based on the ERGs, initiate high pressure SI and open the pressurizer PORVs. This bleed and feed mode of recovery is used for the unlikely situation in which AFW is not available and other modes of secondary cooling (e.g., recovery of main feedwater) cannot be performed. The characteristic pressure associated with this scenario is conservatively assumed to be 2000 psig in WCAP-10319. With capability to open all three pressurizer PORVs at Donald C. Cook Units 1 and 2, the RCS pressure would be expected to be less than 2000 psig for this bleed and feed mode of recovery. Thus, application of the generic curve for this specific contribution is considered to be appropriate.

The other PTS transient scenarios, including those involving loop stagnation (such as secondary depressurization, anticipated transient without scram, and feedline break), do not contribute significantly to the total frequency. This supports the WOG position that the overall risk from PTS is dominated by small LOCA and SGTR events and is not affected by other candidate sequences, including severe cooldown transients that result in stagnant loop conditions. Note that small steamline breaks are no longer significant contributors to the total frequency of flaw extension for a typical Westinghouse PWR as previously suggested in an earlier WOG PTS risk study (Reference 2).

Based on the assessment summarized above, results of the stagnant loop study are considered applicable and conservative for the Donald C. Cook Nuclear Plant. Emergency operating procedures based on the ERGs are available and the operators are trained to follow the procedures. This includes significant actions such as tripping the RCPs based on specified criteria, throttling AFW flow when called for, terminating SI flow when required, and only cooling the RCS within specified limits. Sensitivities to various operator action times and credible failures are also considered, as was noted above for the delayed SI termination SGTR cases.

Based on the results from the stagnant loop evaluation and limiting PTS and screening values as described below, it can be concluded that, as long as plant emergency operating procedures based on the ERGs are followed, stagnant loop transients do not significantly increase the PTS risk for a typical Westinghouse-designed plant such as the Donald C. Cook Nuclear Plant.

3. *Limiting RT_{PTS} Values and Screening Criteria*

In July 1985, the NRC published a new rule under 10 CFR 50.61 entitled "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS) Events." This new rule established screening limits for the calculated reference temperature for pressurized thermal shock (RT_{PTS}) as follows: a) 270°F for plates, forgings, and axial weld materials, and b) 300°F for circumferential weld materials. The RT_{PTS} must be calculated as per paragraph (b)(2) of 10 CFR 50.61.

In May 1991, the NRC issued a revision to 10 CFR 50.61. This revision incorporated the calculational methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Material," U.S. Nuclear Regulatory Commission, May 1988. In addition, this revision required licensees to submit projected values of RT_{PTS} for the reactor beltline materials for the time of the submittal and for the projected expiration date of the operating license. Per this requirement, the Cook Nuclear Plant submitted a plant-specific RT_{PTS} calculation for both units to the NRC (Reference 6). The controlling material and the calculated RT_{PTS} values at the end of the operating licenses are as follows:

Unit 1	Unit 2
Intermediate to lower shell circumferential weld (9-442)	Intermediate shell plate (C5556-2)
Calculated RT_{PTS} = 216°F	Calculated RT_{PTS} = 217°F
Screening Limit = 300°F	Screening Limit = 270°F

The plant-specific RT_{PTS} calculation for Cook Unit 1 is based on the July 1985 version of 10 CFR 50.61 and the plant-specific RT_{PTS} calculation for Cook Unit 2 is based on the May 1991 version of 10 CFR 50.61.

The NRC has approved the submittal and issued a Safety Evaluation report dated October 1, 1991 (Reference 7). The RT_{PTS} will be recalculated and submitted to the NRC whenever changes in core loadings, surveillance measurements, or other information indicates a significant change in projected values per the requirements of 10 CFR 50.61.

4. Emergency Response Guidelines and Their Application for Long Term Cooling

The generic Emergency Response Guidelines (ERGs) developed by the Westinghouse Owners Group contain appropriate steps to prevent or mitigate the effects of PTS events. These generic guidelines were developed as part of the program for implementation of item I.C.1 of NUREG-0737 and were submitted to the NRC (Reference 8). As the ERGs were developed, a PTS review of the ERGs was performed to ensure that the actions taken were appropriate (Reference 9). Revision 1 of the ERGs included the results of this PTS review and this version of the ERGs has been transmitted to the NRC (Reference 10). As reported above, both Donald C. Cook units have limiting RT_{PTS} projected values that are greater than 200°F but less than 250°F. With these results, both units are considered to be in Category II with respect to PTS mitigation actions based on the ERGs (Reference 11). The Donald C. Cook Units 1 and 2 emergency operating procedures are based on the WOG ERGs.

For initial cooldown following either of the dominant accident scenarios, the operator is instructed to use the intact steam generators (S/Gs) to remove plant decay and stored heat. This is done by feeding the S/Gs with auxiliary feedwater to maintain an indicated S/G water level within the narrow-range level instrument span, and relieving S/G pressure by means of the steam dump valves (if offsite power is available or can be quickly restored). The main steam system atmospheric safety or relief valves can be used if the steam dump system is not available. Long term cooling continues using the intact S/Gs, auxiliary feedwater and the atmospheric safety or relief valves.

In order to assure effective long-term cooling for a LOCA certain additional operator actions are assumed. These actions are principally

1. to switch the ECCS from the injection phase to the recirculation phase,
2. to place the reactor coolant pumps in a condition where they can most effectively aid core cooling, and
3. to switch the ECCS from cold leg recirculation to hot leg recirculation at the appropriate time to prevent boron precipitation.

All of these items and other appropriate actions that need to be taken to ensure PTS risk is minimized are specified in the plant emergency operating procedures.

Control of long term cooling for these and other accident scenarios requires specific actions and responses unique to the type of accident that has occurred. The plant emergency operating procedures provide the operator instructions for controlling long term cooling and minimizing PTS risk during anticipated accidents. The emergency operating procedures cover the range of activities required to take the plant to a safe shutdown condition, identify the parameters that must be monitored, the equipment that must be available, and establish the sequence for performing the required activities. Procedural compliance ensures that activities are completed within time frames required to prevent challenging PTS risk concerns and minimize operator errors. The complete set of emergency operating procedures also account for other unusual conditions such as loss of offsite power, some multiple failures (such as LOCA plus SGTR) and equipment failures/unavailability.

14.3.7.1 References for Section 14.3.7

1. Letter No. OG-58 from Mr. R. W. Jurgensen, Chairman, WOG, to Mr. D. G. Eisenhut, Director, NRC, entitled, "Thermal Shock to Reactor Pressure Vessel," dated May 14, 1981. (Attachment "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants" (WCAP-10019)).

2. Letter No. OG-70 from Mr. O. D. Kingsley, Chairman, WOG, to Mr. Harold R. Denton, Director, NRC, entitled, "Supplemental Information on Reactor Vessel Integrity," dated May 18, 1982. (Attachment "Summary of Evaluations Related to Reactor Vessel Integrity" (WCAP-10019-S1)).

3. Letter from Mr. T. M. Novak, NRC, Assistant Director for Operating Reactors, to Mr. O. D. Kingsley, Chairman, WOG, dated March 16, 1982.

4. NRC Policy Issue, Enclosure A, "NRC Staff Evaluation of Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

5. Westinghouse Electric Corporation, "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, From Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants, WCAP-10319, December 1983.

6. Letter No. AEP:NRC:0561D, "Donald C. Cook Nuclear Plant, Units 1 and 2, Updated Reference Temperature, Pressurized Thermal Shock Analyses," dated August 7, 1990, from Mr. P. Alexich to T. E. Murley.

7. NRC SER, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 157 to Facility Operating License No. DPR-58, and Amendment No. 141 to Facility Operating License No. DPR-74, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Units Nos. 1 and 2, Docket Nos. 50-315 and 50-316," dated October 1, 1991.

8. Letter No. OG-64 from Mr. R. W. Jurgensen, Chairman, WOG, to Mr. D. G. Eisenhut, Director, Division of Licensing, entitled, "Emergency Response Guideline Program," dated November 30, 1981.

9. Letter No. OG-72 from Mr. O. D. Kingsley, Chairman, WOG, to Mr. Harold R. Denton, Director, NRC, entitled, "PTS Review of the ERGs," dated June 22, 1982.

10. Letter No. OG-111 from Mr. J. J. Sheppard, Chairman, WOG, to Mr. H. L. Thompson, Director, Division of Human Factors Safety, entitled "Transmittal of Revision 1 of Emergency Response Guidelines Revision 1," dated November 30, 1983.

11. Emergency Response Guidelines - Revision 1B (High-Pressure Version), Westinghouse Owners Group, February 28, 1992.

14.3.8 Nitrogen Blanketing

This section was a portion of the response to question 212.34 of Appendix Q of the original FSAR which addressed the issue of nitrogen blanketing from the accumulators in SBLOCA.

Subsequent to 1977, extensive analyses have been performed to study the general behavior of SBLOCA. Credible sources of non-condensables in SBLOCA have been considered in both references 1 and 2. Accumulator nitrogen was not identified as a potential source of non-condensables. Credible non-condensables were specifically addressed in the development of the NOTRUMP code which is currently the analysis of record for both units.

14.3.9 Containment and Recirculation Sump Analyses

See discussion on this subject in the Unit 1 UFSAR.

References:

1. WCAP 9600, Report on Small Break Accidents for Westinghouse NSSS System, Approved T. M. Anderson, June 1979.

2. WCAP 10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, N. Lee, S. D. Rupprecht, W. R. Schwarz, W. D. Tauche, August 1985.

3. Letter, Steven A. Varga, NRC Staff to John Dolan, May 23, 1985.

CONTAINMENT REGION DESIGNATIONS/IGNITER ASSEMBLY LOCATIONS

Region	Train A Compartment/Area-Elev.	No.	Region	Train B Compartment/Area-Elev.	No.
1	Upper Volume Dome Area -760'	A-29	1	Upper Volume Dome Area -760'	B-29
1	Upper Volume Dome Area -760'	A-30	1	Upper Volume Dome Area -760'	B-30
1	Upper Volume Dome Area -760'	A-31	1	Upper Volume Dome Area -760'	B-31
2	Upper Volume Dome Area -748'	A-32	2	Upper Volume Dome Area -748'	B-32
2	Upper Volume Dome Area -748'	A-33	2	Upper Volume Dome Area -748'	B-33
2	Upper Volume Dome Area -748'	A-34	2	Upper Volume Dome Area -748'	B-34
3	Ice Cond. Upper Plenum-708'	A-1	3	Ice Cond. Upper Plenum-709'	B-1
3	Ice Cond. Upper Plenum-709'	A-2	3	Ice Cond. Upper Plenum-709'	B-2
4	Ice Cond. Upper Plenum-709'	A-3	4	Ice Cond. Upper Plenum-709'	B-3
4	Ice Cond. Upper Plenum-709'	A-4	4	Ice Cond. Upper Plenum-709'	B-4
4	Ice Cond. Upper Plenum-709'	A-5	4	Ice Cond. Upper Plenum-709'	B-5
5	Ice Cond. Upper Plenum-710'	A-6	5	Ice Cond. Upper Plenum-709'	B-6
5	Ice Cond. Upper Plenum-709'	A-7	5	Ice Cond. Upper Plenum-709'	B-7
6	Outside #2 SG Enclosure-662'	A-14	6	Outside #2 SG Enclosure-659'	B-14
6	Outside #3 SG Enclosure-662'	A-15	6	Outside #3 SG Enclosure-659'	B-15
7	Outside #1 SG Enclosure-659'	A-13	7	Outside #1 SG Enclosure-662'	B-13
7	Outside #4 SG Enclosure-662'	A-16	7	Outside #4 SG Enclosure-659'	B-16
7	Outside PZR Enclosure-662'	A-17	7	Outside PZR Enclosure-659'	B-17
8	East Fan/Accumulator Room-631'	A-24	8	East Fan/Accumulator Room-630'	B-24
8	East Fan/Accumulator Room-629'	A-25	8	East Fan/Accumulator Room-629'	B-25
9	West Fan/Accumulator Room-629'	A-26	9	West Fan/Accumulator Room-623'	B-26
9	West Fan/Accumulator Room-634'	A-27	9	West Fan/Accumulator Room-634'	B-27
10	Instrument Room-620'	A-35	10	Instrument Room-620'	B-35
11	Primary Shield Wall-647'	A-18	11	Primary Shield Wall-642'	B-18
11	Primary Shield Wall-648'	A-19	11	Primary Shield Wall-637'	B-19
11	Primary Shield Wall-648'	A-20	11	Primary Shield Wall-636'	B-20
11	Primary Shield Wall-648'	A-21	11	Primary Shield Wall-636'	B-21
11	Primary Shield Wall-641'	A-22	11	Primary Shield Wall-637'	B-22

Region	Train A Compartment/Area-Elev.	No.	Region	Train B Compartment/Area-Elev.	No.
11	Primary Shield Wall-648'	A-23	11	Primary Shield Wall-645'	B-23
12	Inside #1 SG Enclosure-686'	A-8	12	Inside #1 SG Enclosure-686'	B-8
13	Inside #2 SG Enclosure-686'	A-9	13	Inside #2 SG Enclosure-686'	B-9
14	Inside #3 SG Enclosure-686'	A-10	14	Inside #3 SG Enclosure-686'	B-10
15	Inside #4 SG Enclosure-686'	A-11	15	Inside #4 SG Enclosure-685'	B-11
16	Inside PZR Enclosure-686'	A-12	16	Inside PZR Enclosure-682'	B-12
17	Vicinity of PRT-618'	A-28	17	Vicinity of PRT-618'	B-28

Key:

SG - Steam Generator
PZR - Pressurizer
PRT - Pressurizer Relief Tank

Note: The locations, igniter identification numbers, and containment region designations given are for Cook Nuclear Plant Unit 2 and are similar for Unit 1.

Figure 14.3.7-1

FREQUENCY OF SIGNIFICANT FLAW EXTENSION

FOR LONGITUDINAL FLAWS IN A TYPICAL WESTINGHOUSE PWR