TABLE 15.4-1
(Sheet 1 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
a. Uncontrolled RCCA Bank withdrawal from a Subcritical or Low-Low Power Startup Condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	10.9
	Minimum DNBR occurs	12.3
	Peak heat flux occurs	12.3
	Peak average clad temperature occurs	12.3
	Peak average fuel temperature occurs	12.6

TABLE 15.4-1
(Sheet 2 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
b. Uncontrolled RCCA Bank Withdrawal at Power (minimum feedback)		
Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0.0
	Power range high neutron flux high setpoint reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.1
Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0.0
	Overtemperature ΔT reactor trip setpoint reached	60.0
	Rods begin to fall into core	62.5
	Minimum DNBR occurs	63.1
c. CVCS Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant		
1. Dilution during refueling	Dilution begins	0
	Shutdown Monitor alarm occurs	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 1800

TABLE 15.4-1
(Sheet 3 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
2. Dilution during cold shutdown (filled loops)	Dilution begins	0
	Shutdown Monitor alarm occurs	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
3. Dilution during cold shutdown (drained loops)	Dilution begins	0
	Shutdown Monitor alarm occurs	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
4. Dilution during hot shutdown	Dilution begins	0
	Shutdown Monitor alarm occurs	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
5. Dilution during hot standby	Dilution begins	0
	Shutdown Monitor alarm occurs	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900

TABLE 15.4-1
(Sheet 4 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
6. Dilution during startup	Dilution begins	0
	Source range high neutron flux trip/alarm	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
7. Dilution during power operation		
(a) Automatic reactor control	Dilution begins	0
	Rod insertion limit alarms occur at	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
(b) Manual reactor control	Dilution begins	0
	Rod insertion limit alarms, or OTΔT or other trip alarm at	T
	Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900

TABLE 15.4-1
(Sheet 5 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
d. Rod Cluster Control Assembly Ejection		
1. Beginning-of-Life Full Power	Initiation of rod ejection	0.0
	Power range high neutron flux setpoint reached	0.05
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.00
	Peak clad temperature occurs	2.07
	Peak heat flux occurs	2.09
2. End-of-Life Zero Power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.19
	Peak nuclear power occurs	0.23
	Rods begin to fall into core	0.69
	Peak heat flux occurs	1.42
	Peak clad temperature occurs	1.43
	Peak fuel average temperature occurs	1.67

TABLE 15.4-2

PARAMETERS USED IN THE RCCA EJECTION ACCIDENT

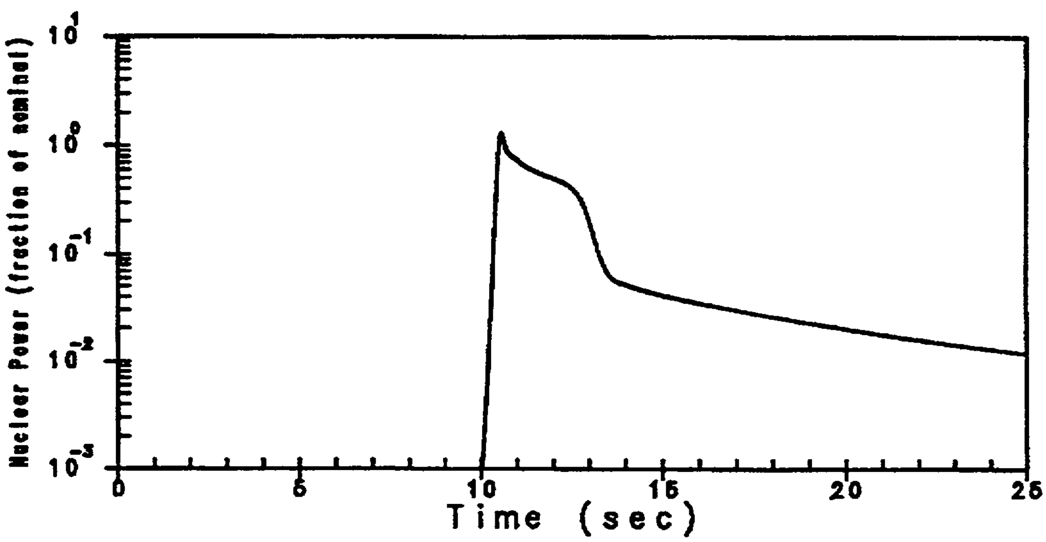
Time in Life - Power:	BOL-HFP	BOL-HZP	EOL-HFP	EOL-HZP
Power Level (%)	102	0	102	0
Ejected Rod Worth (%$\Delta\rho$)	0.25	0.78	0.25	0.85
Delayed Neutron Fraction (%)	0.54	0.54	0.44	0.44
Feedback Reactivity Weighting	1.30	2.071	1.30	3.55
Trip Reactivity (%$\Delta\rho$)	4.0	2.0	4.0	2.0
F_Q Before Rod Ejection	2.5	-	2.5	-
F_Q After Rod Ejection	6.0	11.5	7.0	26.0
Number of Operation Pumps	4	2	4	2
Max. Fuel C/L Temperature (°F)	*	3750	4575	3878
Max. Fuel Avg. Temperature (°F)	3846	3258	3377	3641
Max. Fuel Stored Energy (cal/gm) [Btu/lb]	167 [299.5]	137 [245.9]	143 [256.5]	147 [263.9]
Fuel Melt (%)	1.2	0	0	0

* less than 10% fuel melt

TABLE 15.4-3
(Sheet 1 of 2)

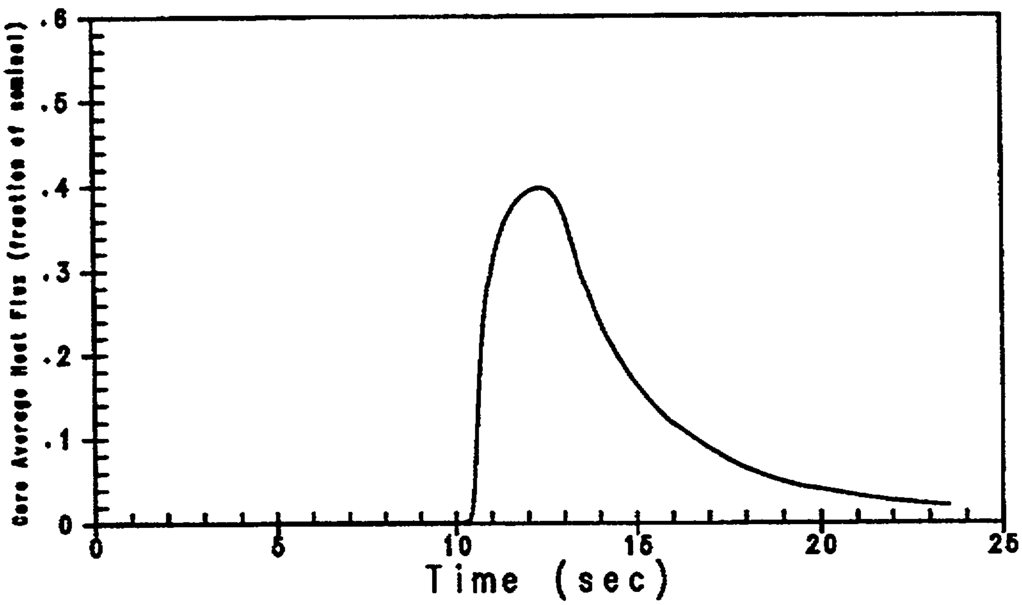
SUMMARY OF PARAMETERS AND ASSUMPTIONS
USED FOR THE ROD EJECTION ACCIDENT

			Conservative Analysis	Realistic Analysis
I.	Data and assumptions used to estimate radioactive source from postulated accident			
	A.	Power level	Appendix 15B	Appendix 15B
	B.	Burnup	Appendix 15B	Appendix 15B
	C.	Percent of fuel perforated	Subsection 15.4.8.3	Subsection 15.4.8.3
	D.	Release of activity by nuclide	Table 15.4-4	Table 15.4-9
	E.	Iodine fractions (elemental, organic and particulate)	Appendix 15B	Appendix 15B
	F.	Reactor coolant and secondary coolant activity before the accident	Subsection 15.4.8.3	Subsection 15.4.8.3
II.	Data and assumptions used to estimate activity released			
	A.	Primary containment leak rate	Appendix 15B	Appendix 15B
	B.	Secondary containment leak rate	Appendix 15B	Appendix 15B
	C.	Valve movement times	N/A	N/A
	D.	Adsorption and filtration efficiency	Appendix 15B	Appendix 15B
	E.	Recirculation system parameters (flow rates versus time, missing factor, etc.)	N/A	N/A



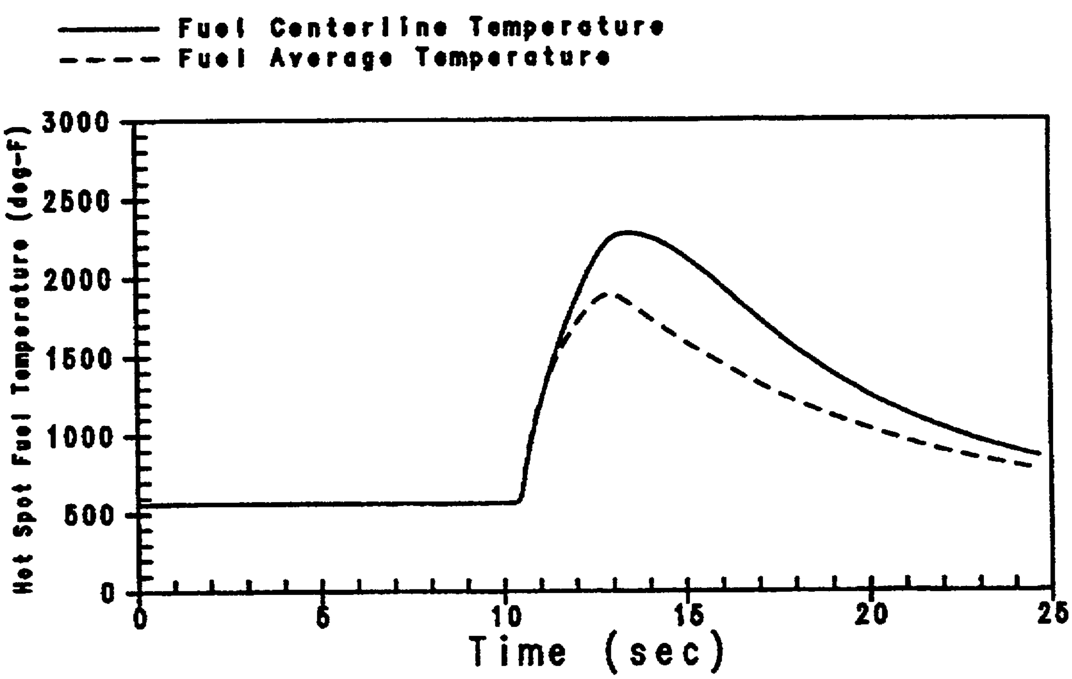
G:\Word\Images_P\Images_P.UFSAR\51141.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition	
	REV. 07	FIGURE 15.4-1



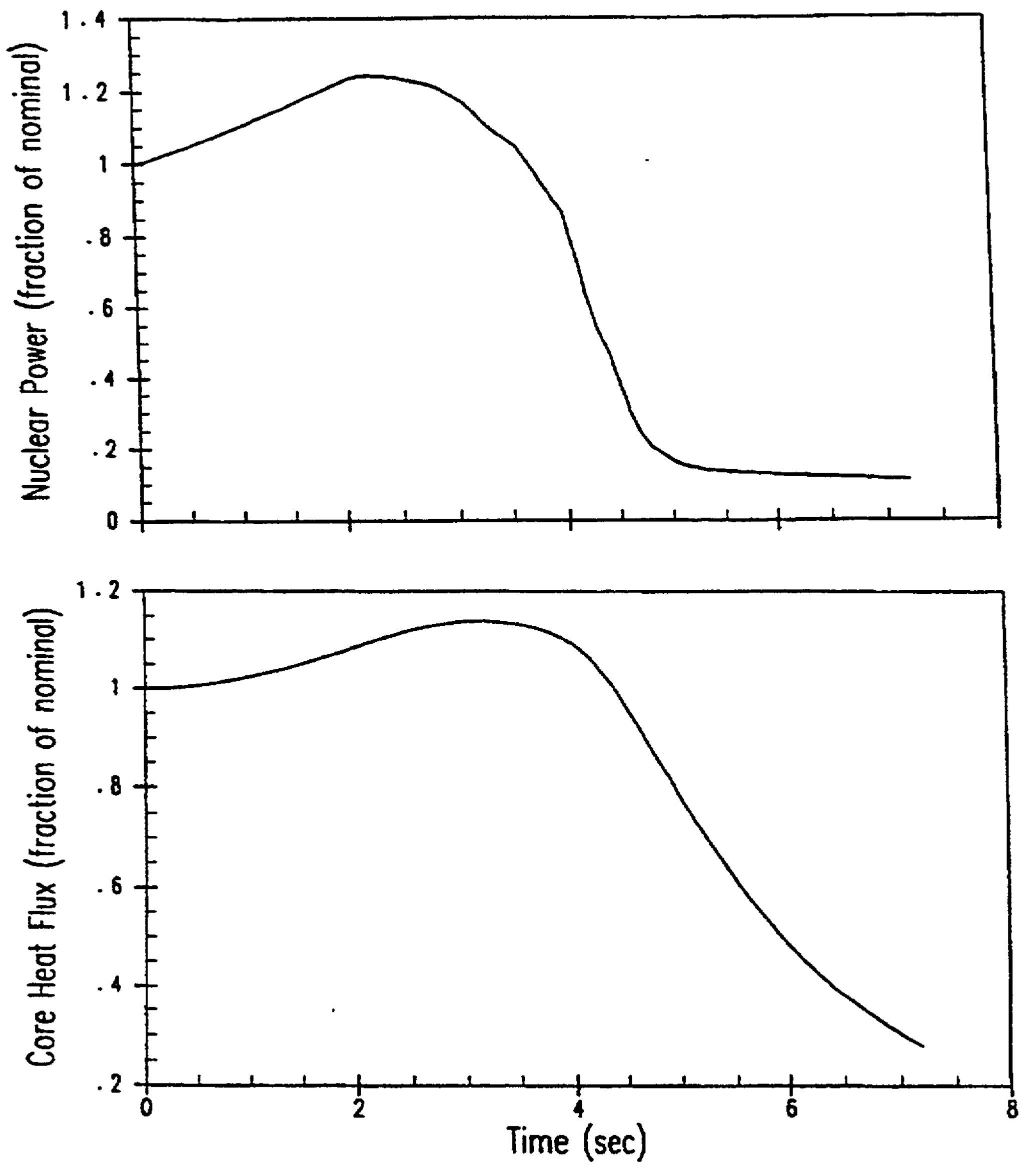
G:\Word\Images_P\Images_P.UFSAR\51142.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Heat Flux Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition	
	REV. 07	FIGURE 15.4-2



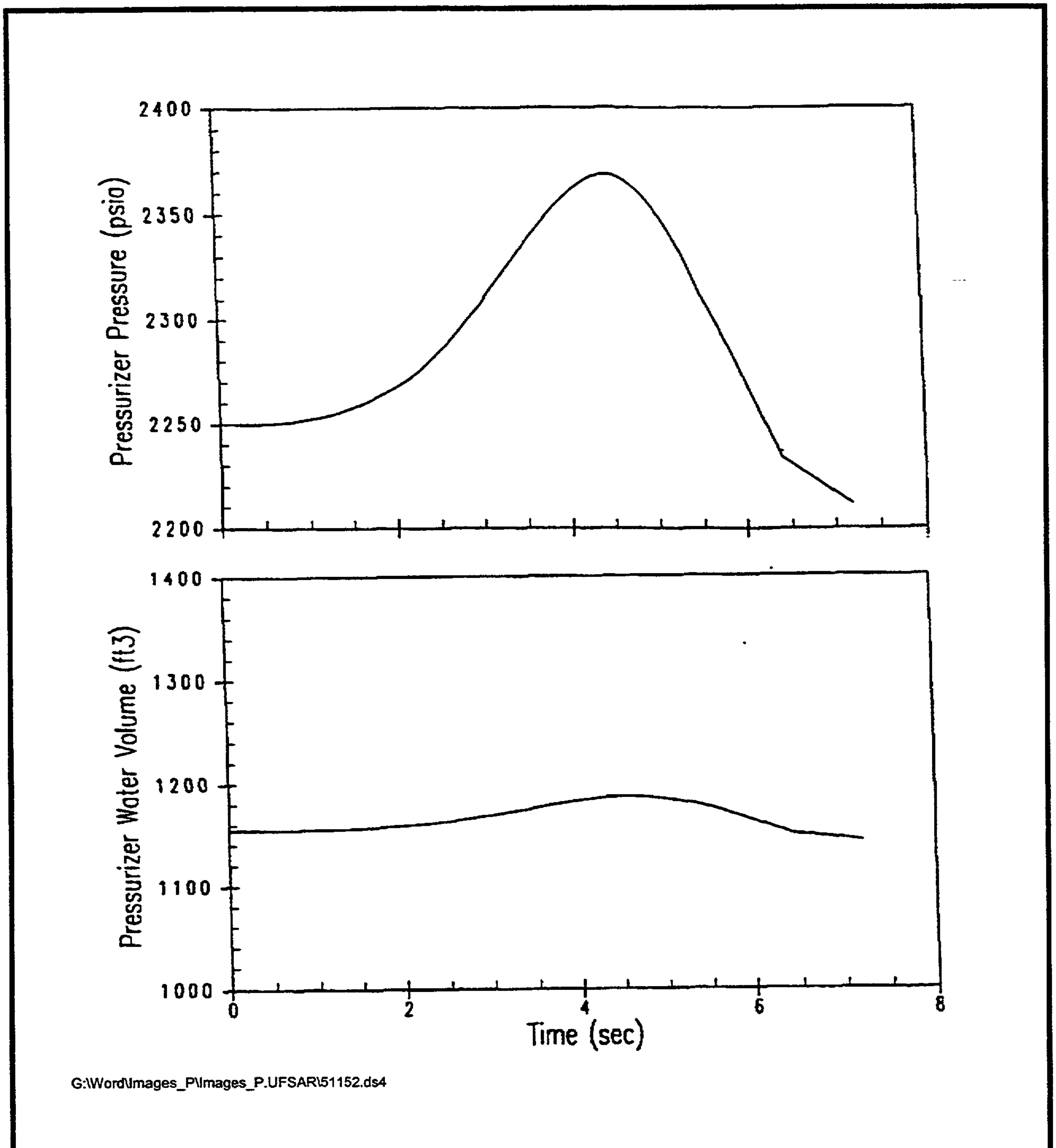
G:\Word\Images_P\Images_P.UFSAR\51143.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline and Fuel Average Hot Spot Temperature Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition	
	REV. 07	FIGURE 15.4-3

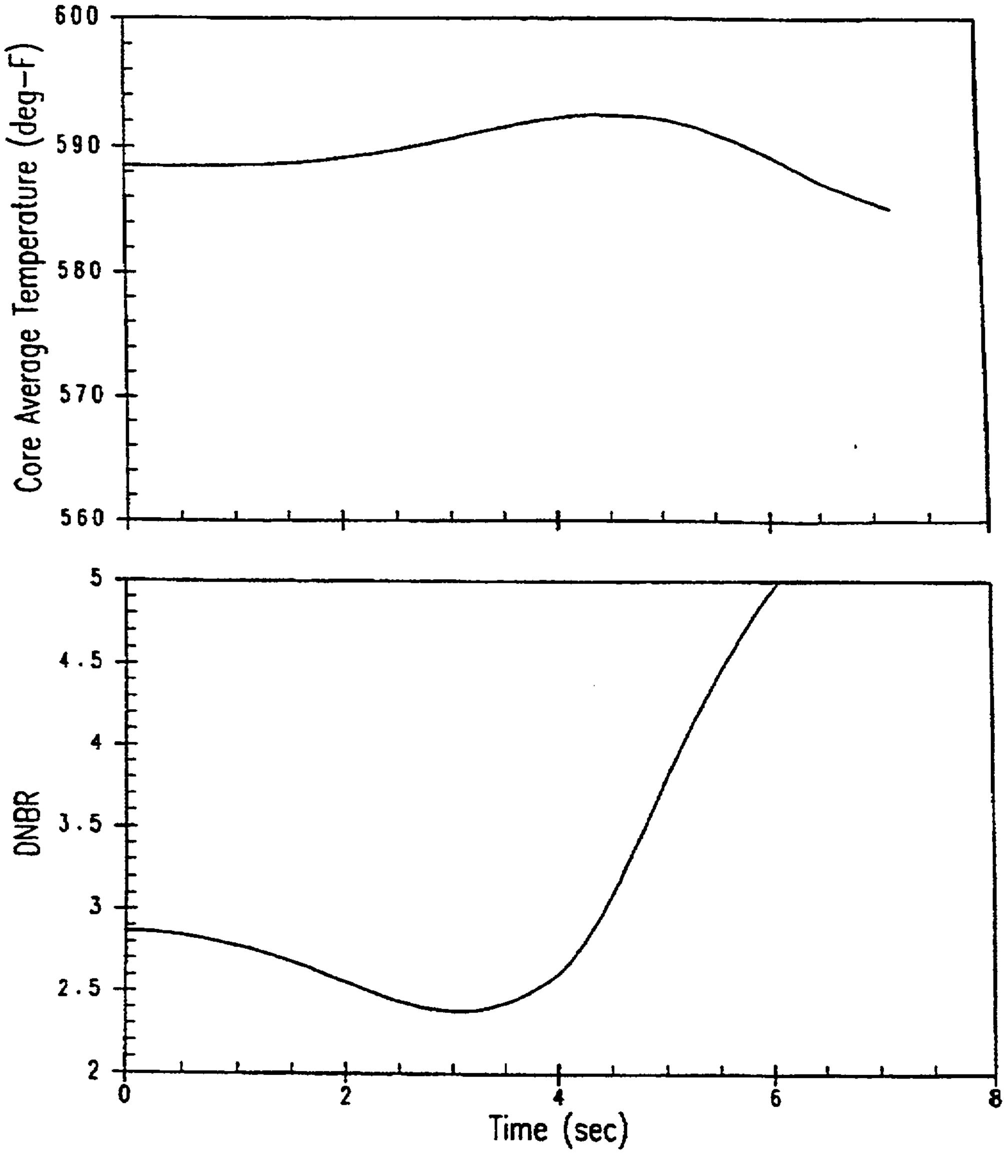


G:\Word\Images_P\Images_P.UFSAR\51151.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-4, Sh 1

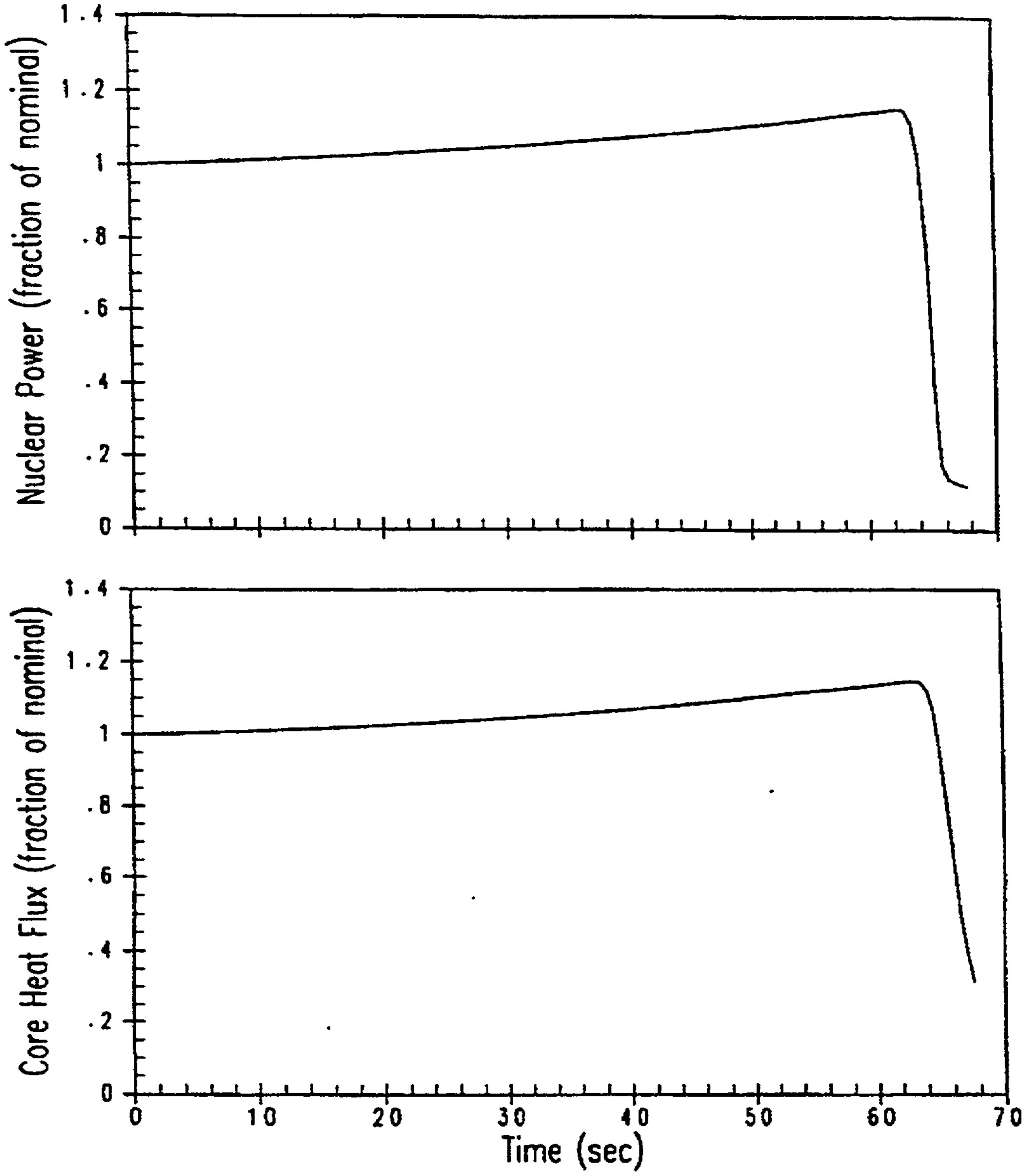


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-4, Sh 2



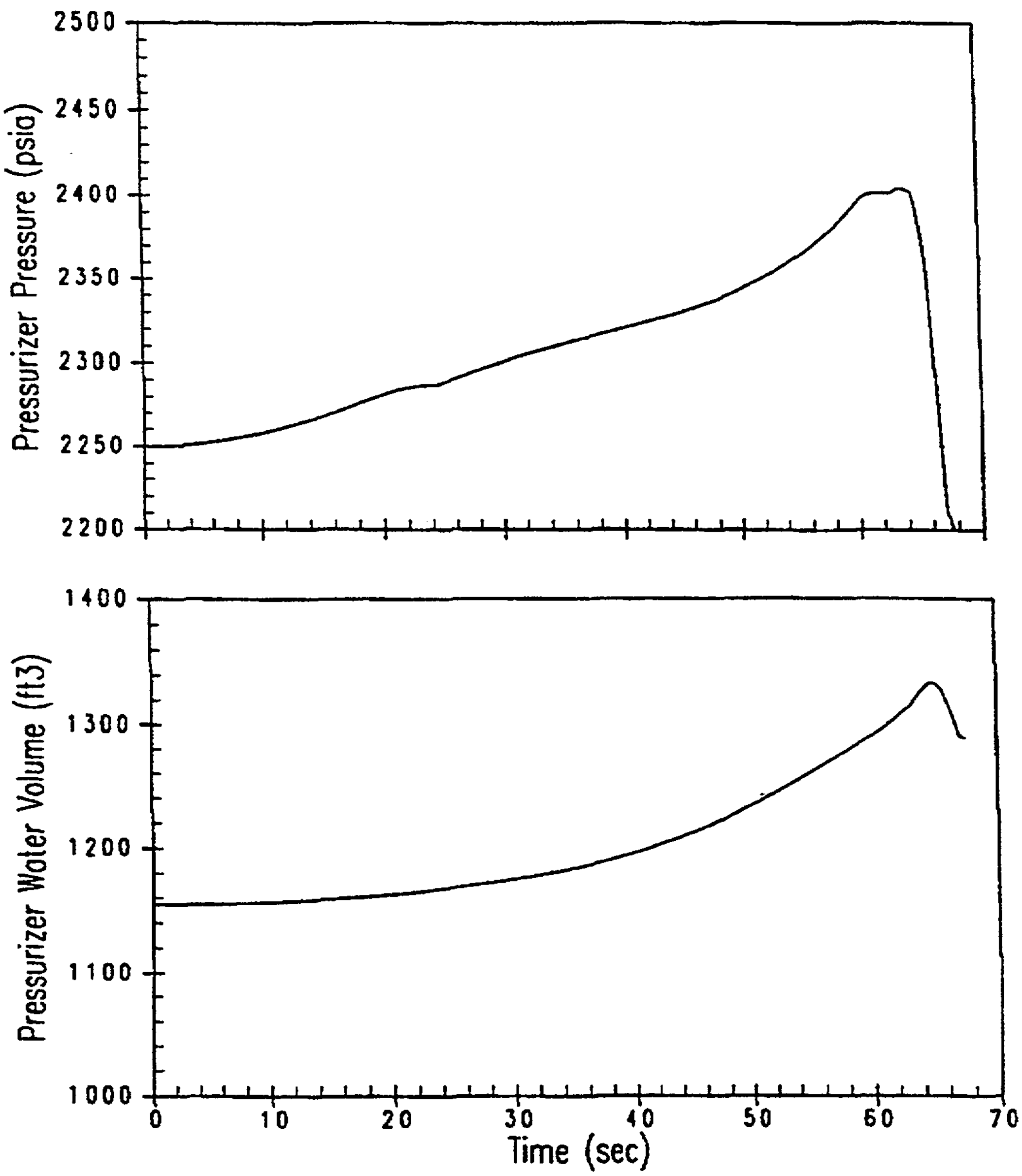
G:\Word\Images_P\Images_P.UFSAR\51153.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and DNBR Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-4, Sh 3



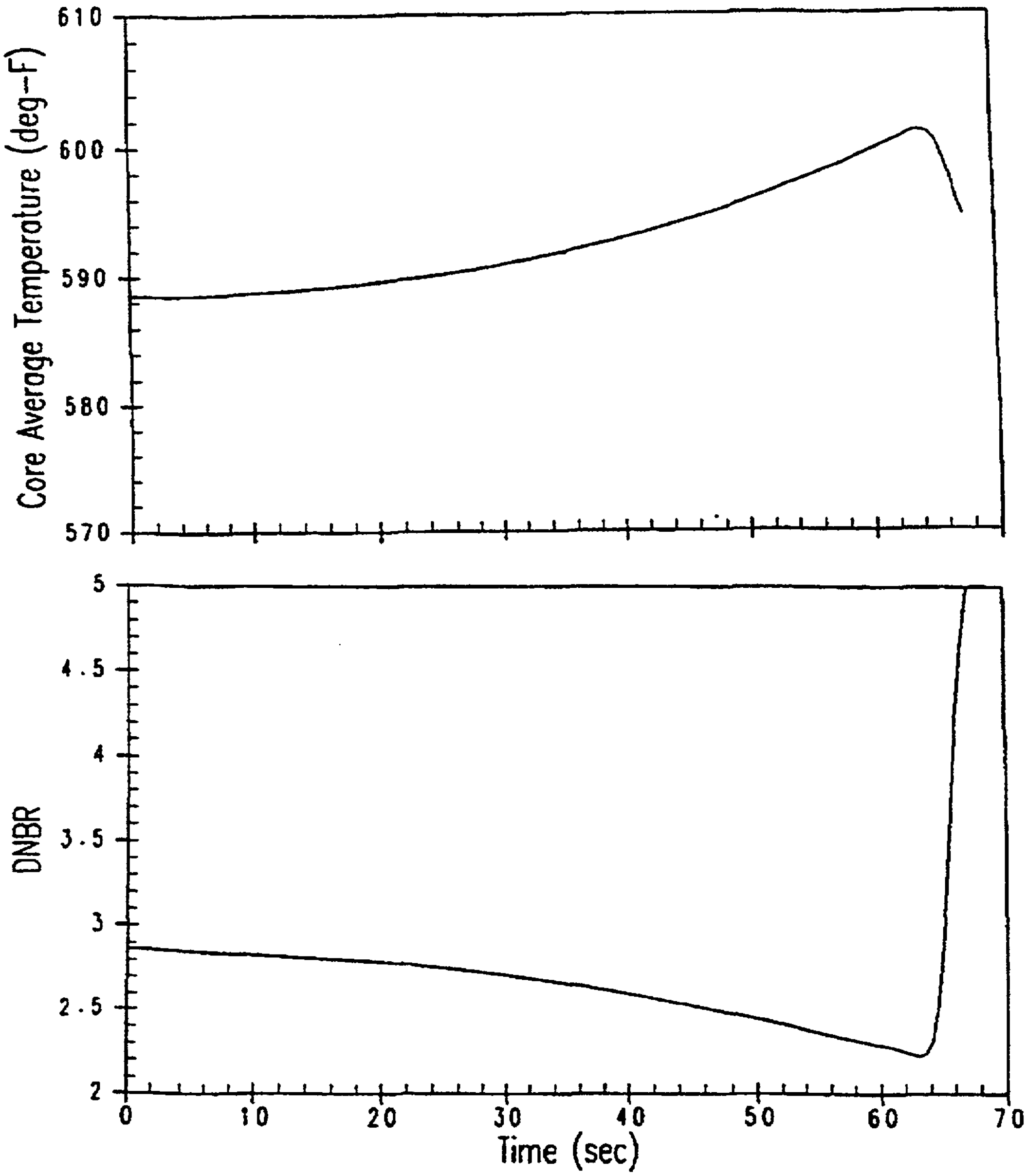
G:\Word\Images_P\Images_P.UFSAR\51154.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-5, Sh 1



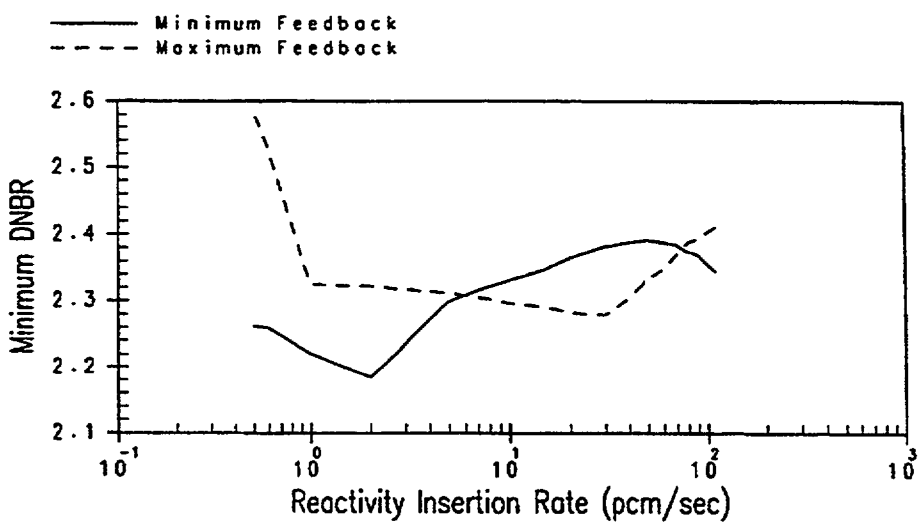
G:\Word\Images_P\Images_P.UFSAR\51155.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-5, SH. 2



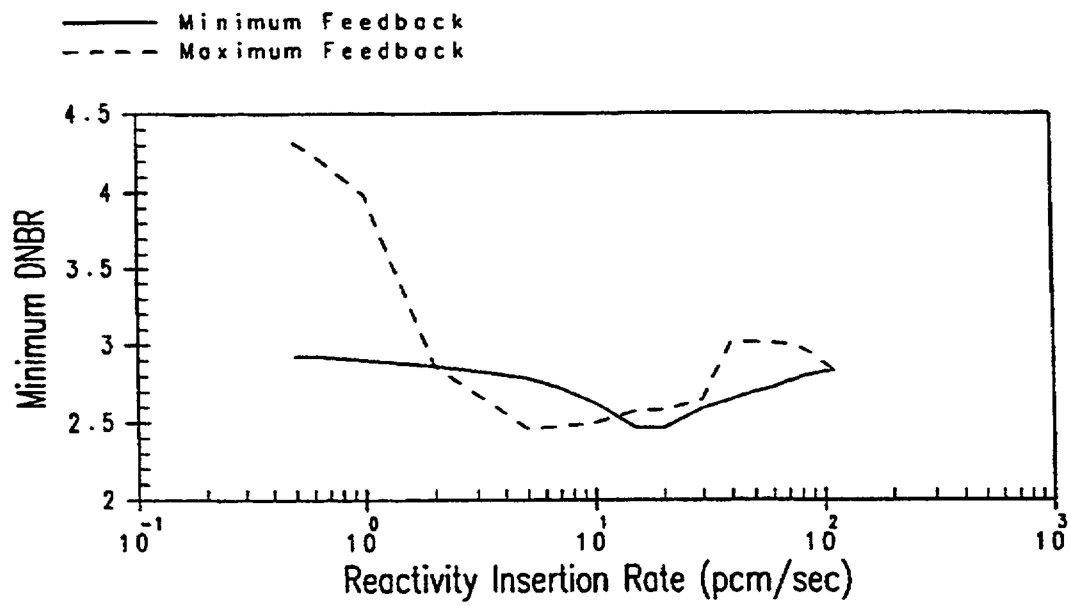
G:\Word\Images_P\Images_P.UFSAR\51156.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and DNBR Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
	REV. 07	FIGURE 15.4-5 SH. 3



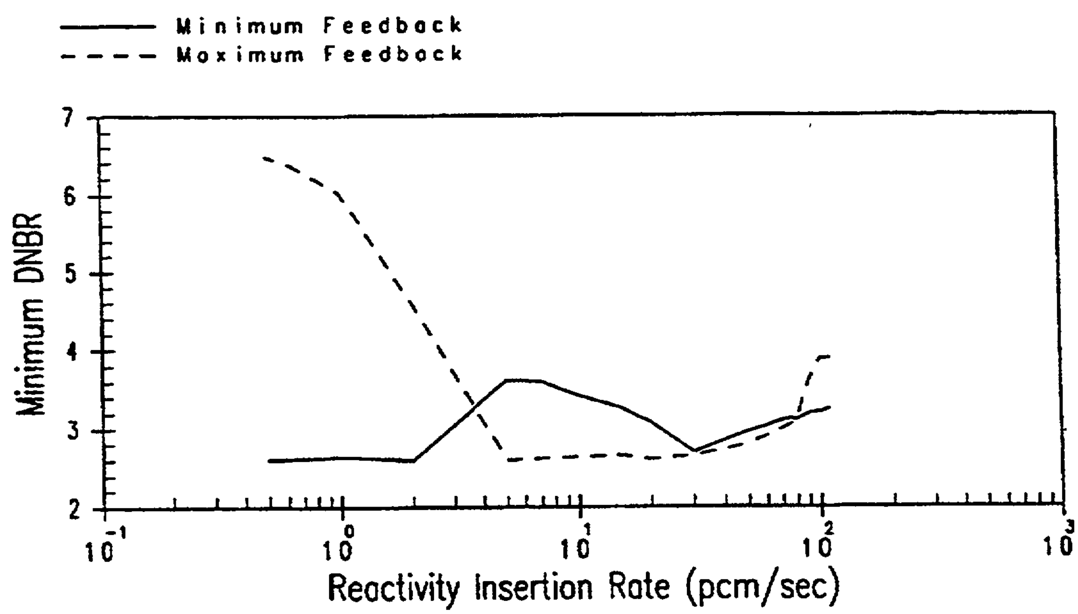
G:\Word\Images_P\Images_P.UFSAR\51157.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled Rcca Bank Withdrawal at 100% Power	
	REV. 07	FIGURE 15.4-6



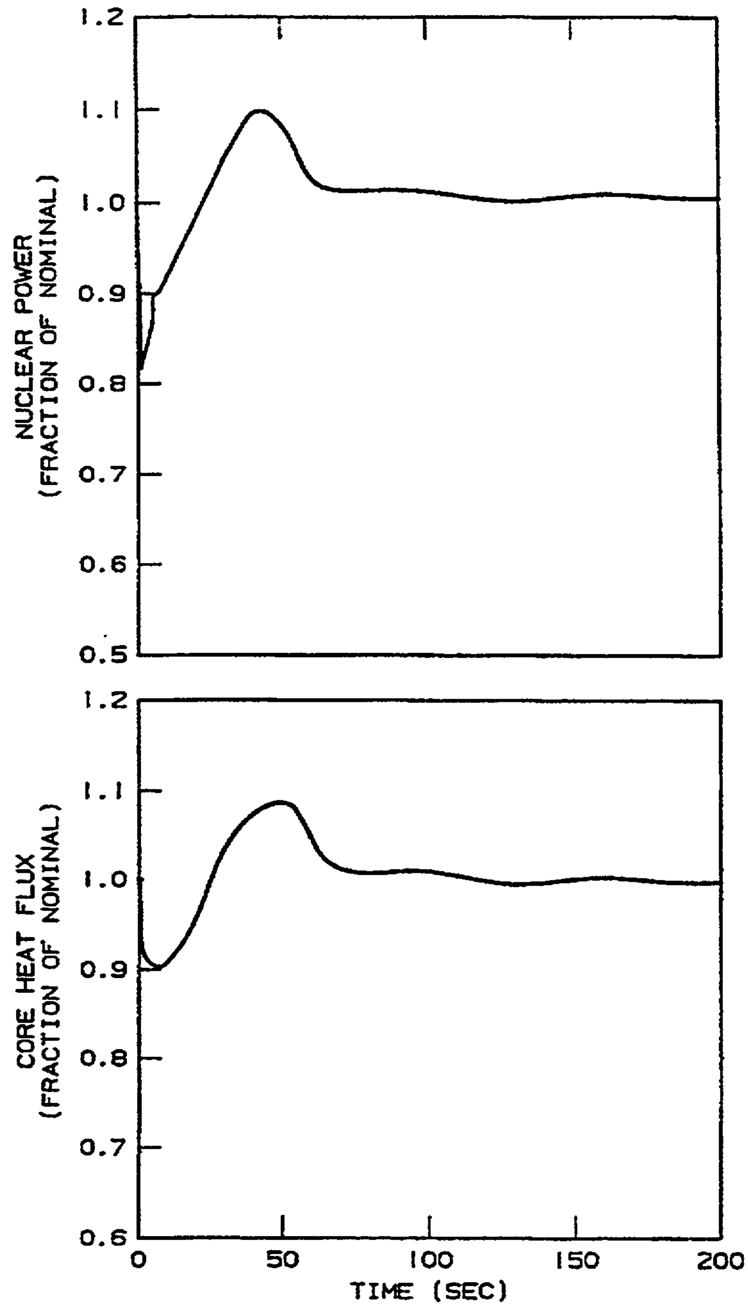
G:\Word\Images_P\Images_P.UFSAR\51158.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 60% Power	
	REV. 07	FIGURE 15.4-7



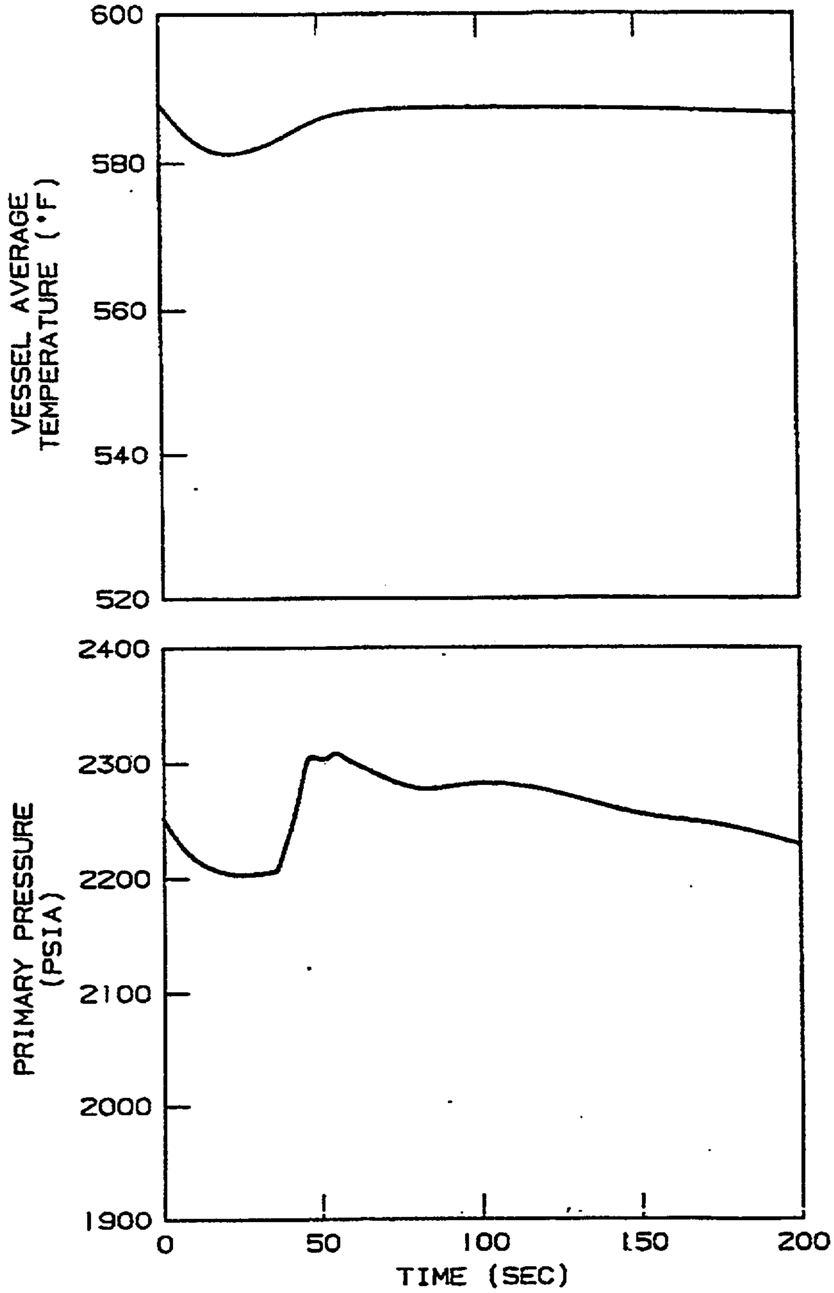
G:\Word\Images_P\Images_P.UFSAR\51159.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 10% Power	
	REV. 07	FIGURE 15.4-8



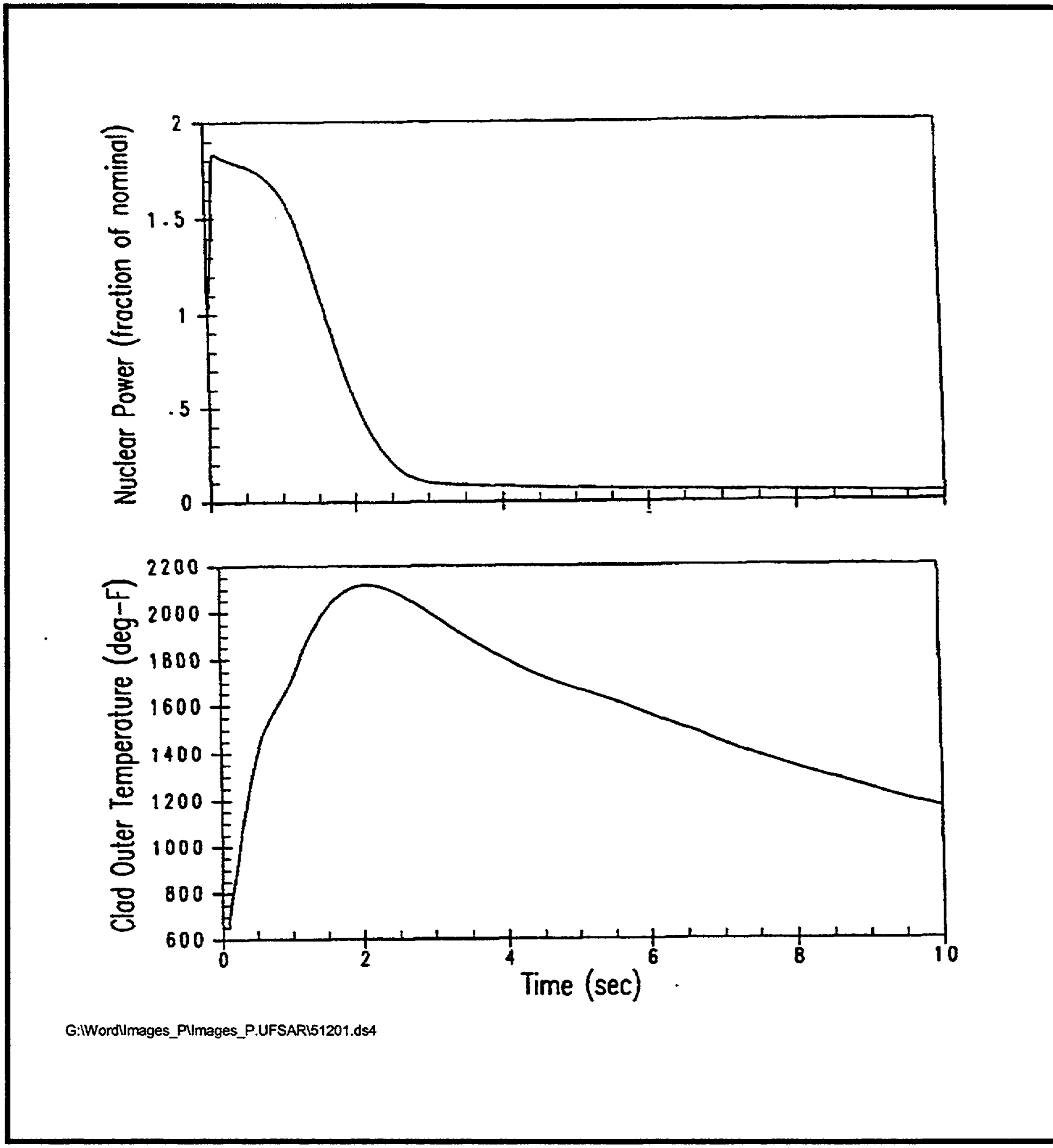
G:\Word\Images_P\Images_P.UFSAR\51161.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for a Dropped RCCA	
	REV. 07	FIGURE 15.4-9 SH.1

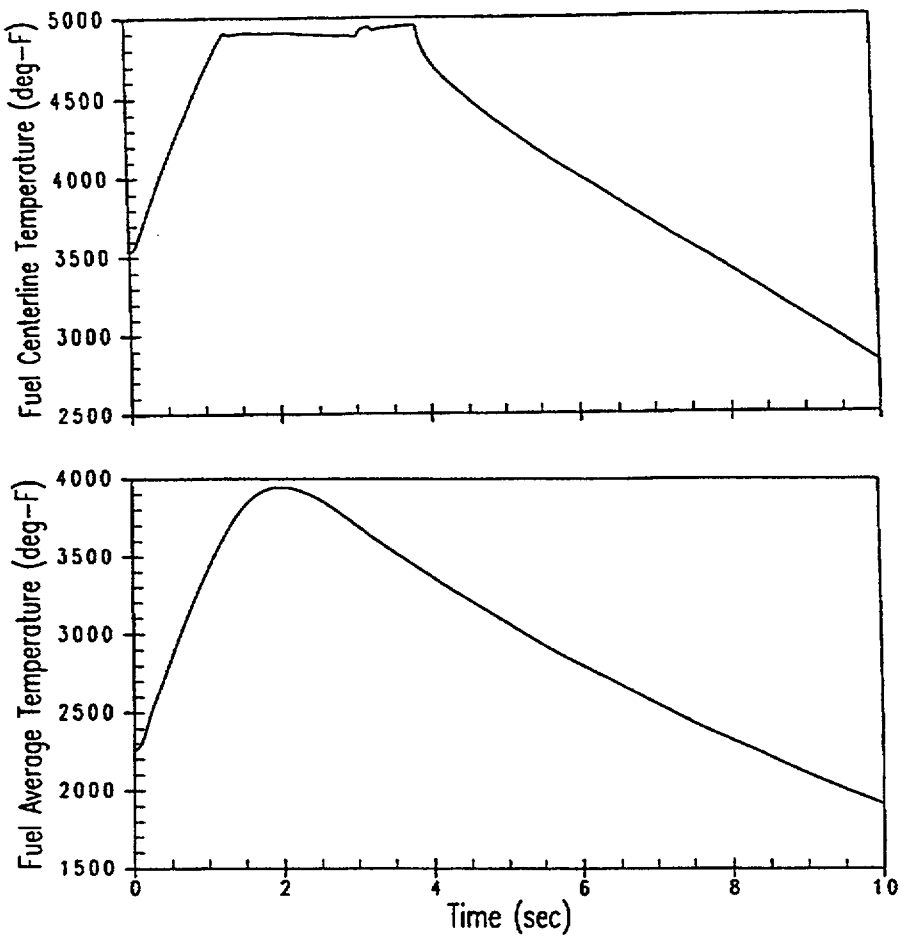


G:\Word\Images_P\Images_P.UFSAR\51162.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Average Coolant Temperature and Pressurizer Pressure for a Dropped RCCA	
	REV. 07	FIGURE 15.4-9 SH. 2

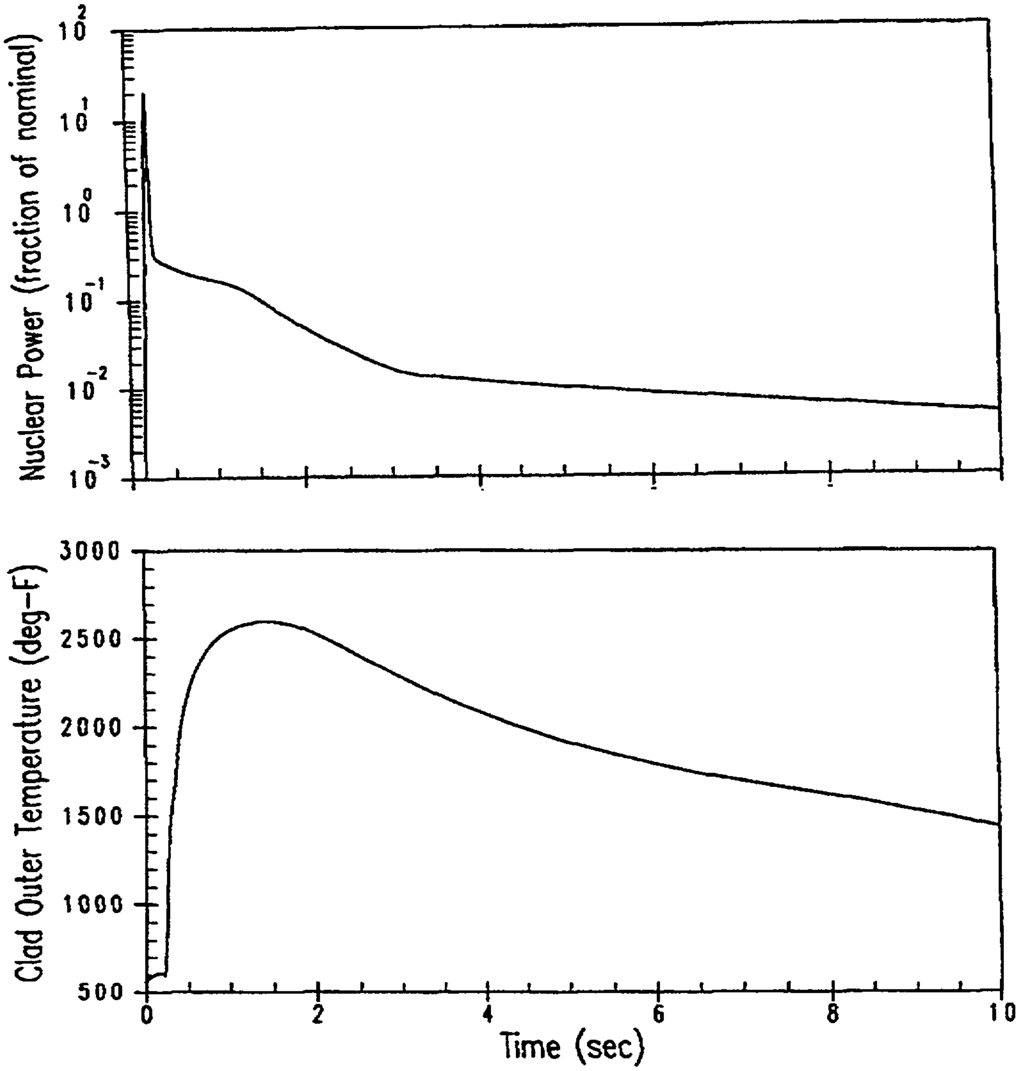


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Clad Outer Temperature Transients for an Uncontrolled RCCA Ejection at BOL HFP	
	REV. 07	FIGURE 15.4-10 SH. 1



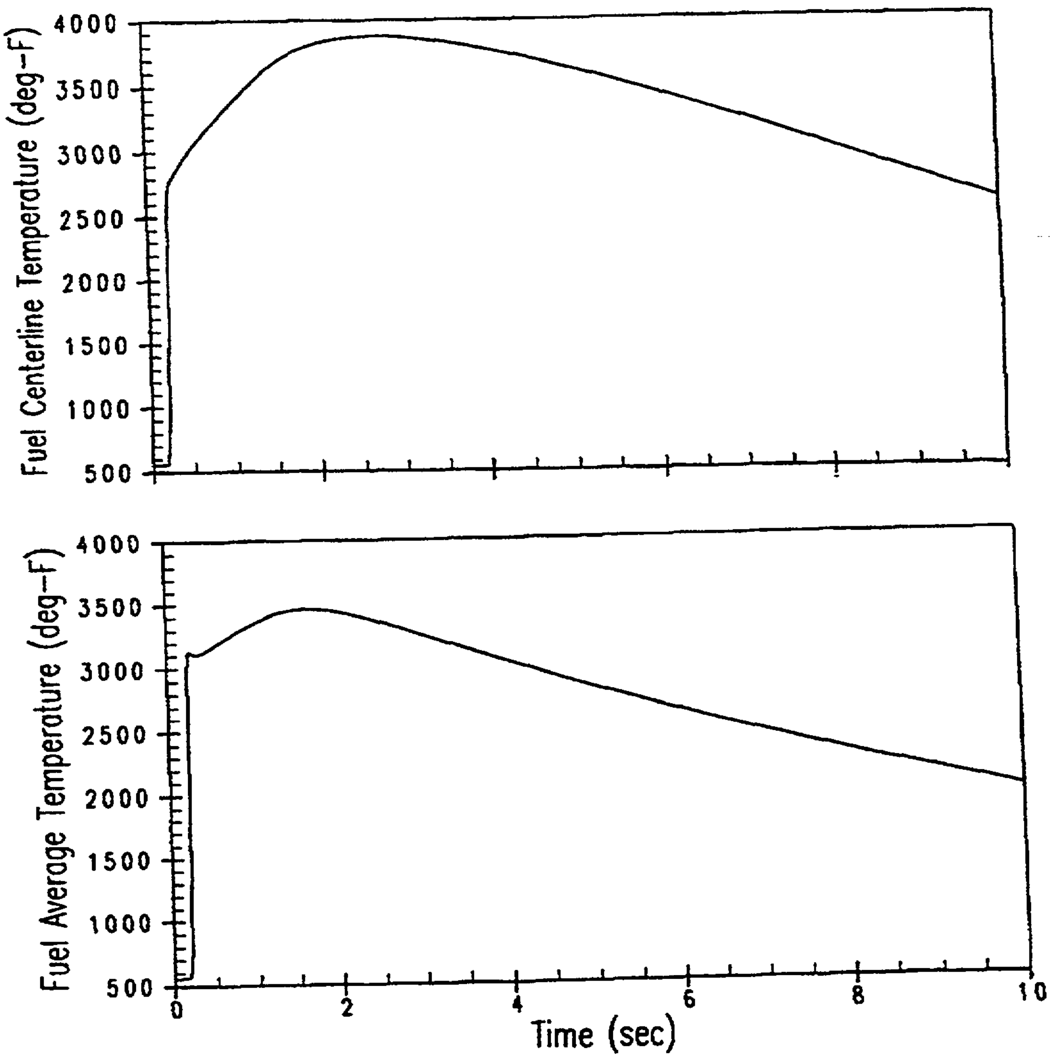
G:\Word\Images_P\Images_P.UFSAR\51202.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline Temperature and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at BOL HFP	
	REV. 07	FIGURE 15.4-10 SH. 2



G:\Word\Images_P\Images_P.UFSAR\51203.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Clad Outer Temperature Transients for an Uncontrolled RCCA Ejection at EOL HZP	
	REV. 07	FIGURE 15.4-11 SH. 1



G:\Word\Images_P\Images_P.UFSAR\51204.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at COL HZP	
	REV. 07	FIGURE 15.4-11 SH. 2

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

a. Inadvertent operation of Emergency Core Cooling System during power operation

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Subsection 15.0.1 contains a discussion of ANS classifications.

15.5.1 Inadvertent Operation of Emergency Core Cooling System during Power Operation

15.5.1.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank. The charging pumps then force highly concentrated (between 2700 and 2900 ppm) boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the Reactor Coolant System (RCS) is at normal RCS pressure.

A Safety Injection System (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal, the operator should determine if the spurious signal was transient or steady-state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

If the Reactor Protection System does not produce an immediate trip as a result of the spurious SIS signal, the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage, pressurizer pressure and water level drop. Load will decrease due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

The time of trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler and moderator coefficients.

Recovery from this second case is made in the same manner as described for the case where the SIS signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. Since the negative reactivity from the injected boron causes reactor power to decrease, the time at which reactor trip occurs has little effect on DNBR.

A second issue associated with this event is the possibility of the pressurizer overfilling, especially a possible condition where the pressurizer is water-solid and its pressure reaches the setpoint of the pressurizer safety relief valves. In this condition, water would pass through these safety relief valves, which could damage the valves and challenge the ability to ensure the RCS boundary can be isolated. The analysis focuses on the pressurizer filling aspects of the event and demonstrates there is no water flow through these pressurizer safety relief valves.

This event is classified as a Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The spurious operation of the ECCS is analyzed by employing the detailed digital computer program LOFTRAN[1]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the ECCS. The program computes pertinent plant variables including temperatures, pressures, pressurizer water level and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits for DNBR.

The analysis employs several plant parameters at maximum and minimum values to maximize the rate for pressurizer filling. For example, the analysis employs maximum reactivity feedback, maximum initial pressurizer water level, and the minimum initial reactor coolant temperature to maximize the rate of pressurizer filling.

Additional analysis of this event has been performed using RETRAN. The RETRAN computer code is discussed in UFSAR Section 15.0.11.

The assumptions are as follows:

1. Initial Operating Conditions

   Initial reactor power is assumed to be at the maximum value, initial reactor coolant temperature is assumed to be at the minimum value, initial pressurizer pressure is assumed to be at its minimum value, and the initial pressurizer water level is assumed to be at its maximum value, consistent with steady-state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity

   A least-negative moderator temperature coefficient was used. A low (absolute value) Doppler power coefficient was assumed.

3. Reactor Control

   The reactor was assumed to be in manual control.

4. Pressurizer Control

   Pressurizer heaters and spray are assumed to be operable to increase the rate of pressurizer filling. The pressurizer sprays act to reduce the RCS pressure, thus increasing ECCS injection. The pressurizer heaters act to add energy to the pressurizer fluid, thus increasing the pressurizer fluid volume through thermal expansion.

5. Boron Injection

   At time zero, two charging pumps conservatively inject 2,900 ppm borated water into the cold leg of each loop.

6. Reactor Trip

   The reactor and turbine are assumed to trip upon receipt of the SI signal. Assuming reactor and turbine trip on SI minimizes the heat removal capability of the RCS, thereby maximizing the RCS inventory increase through SI flow and thermal expansion of the RCS fluid.

7. Pressurizer PORVs

   Two cases are analyzed. In the first case, analyzed using LOFTRAN, credit is taken for one pressurizer PORV. In the second case, analyzed using RETRAN, no credit is taken for any pressurizer PORV operation.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

Figure 15.5-1 shows the transient response to inadvertent operation of the ECCS during power operation. The calculated sequence of events is shown on Table 15.5-1.

Figure 15.5-1 depicts the transient response of the pressurizer overfill analysis. The reactor is tripped at the initiation of the event resulting in the reactor power quickly reducing to decay heat loads. The RCS temperature and pressure quickly stabilize following reactor trip. The pressurizer pressure remains at a constant value until the pressurizer reaches a water solid condition. Once the pressurizer becomes water solid, the pressure quickly increases until at least one PORV provides relief. The secondary system steam flow is isolated at the time of reactor trip to minimize primary-to-secondary heat transfer and increase primary fluid thermal expansion.

Table 15.5-1 also shows that a minimum of 10.1 minutes is available for operator action to terminate ECCS and thereby prevent pressurizer safety relief valve (PSRV) challenges without credit for operation of the pressurizer PORVs. This result has been demonstrated using RETRAN. The RETRAN computer code is discussed in UFSAR Section 15.0.11.

A typical DNBR response for the inadvertent ECCS event, illustrating how DNBR increases throughout the event, is shown in Figure 15.5-1, Sheet 3. If the SI signal does not trip the reactor and turbine, then nuclear power would decrease as borated water is added to the core. Since steam flow would be maintained, the mismatch between nuclear power and load would cause Tavg, pressurizer pressure, and pressurizer water volume to decrease until the low pressurizer pressure reactor trip setpoint is reached. The DNBR would increase, due mainly to the decrease in power and Tavg, and always remain above the safety limit value. Therefore, this event would not pose a challenge to fuel clad integrity.

15.5.1.3 Radiological Consequences

No radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.5.1.4 Conclusions

Results of the analysis show that spurious safety injection presents no hazard to the integrity of the Reactor Coolant System.

The DNBR ratio is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the Reactor Coolant System.

Analytical results show there will be no water flow through the pressurizer safety relief valves (PSRVs) as a consequence of inadvertent operation of ECCS during power operation provided that either of two scenarios occurs:

- There is no water flow through the PSRVs provided that at least one PORV is available for relief; or
- A minimum of 10.1 minutes is available for operator action to terminate ECCS and thereby prevent PSRV challenges without credit for operation of the pressurizer PORVs.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Transients due to CVCS malfunctions that increase the reactor coolant inventory can be divided into three categories:

Category 1	CVCS malfunctions that result in the injection of water with a boron concentration greater than the RCS boron concentration.
Category 2	CVCS malfunctions that result in the injection of water with a boron concentration less than the RCS boron concentration.
Category 3	CVCS malfunctions that result in the injection of water with a boron concentration equal to the RCS boron concentration.

There are two possible criteria for evaluating these transients: core integrity and overfilling of the pressurizer. Transients of the type listed in Category 1 are bounded by the "inadvertent operation of emergency core cooling system analysis" presented in Subsection 15.5.1. Transients of the type listed in Category 2 are bounded by the "CVCS malfunction that results in a decrease in boron concentration in the reactor coolant" presented in Subsection 15.4.6.

CVCS malfunctions of the type described under Category 3 will not result in any significant nuclear power or RCS temperature transient. This type of transient may result in filling the pressurizer. An analysis of the CVCS malfunction that results in injection of water with a boron concentration equal to the RCS boron concentration are presented in this section.

CVCS Malfunctions that Result in the Injection of Water with a Boron Concentration Equal to the RCS Boron Concentration

a. Identification of Causes and Accident Description

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow would be isolated. The worst single failure for this event would be another pressurizer level channel failing in an "as is" condition or a low condition. This will defeat the reactor trip on 2 out of 3 high pressurizer level channels. To prevent filling the pressurizer, the operator must be relied upon to terminate charging. This event is classified as a Condition II incident (an incident of moderate frequency).

b. Analysis of Effects and Consequences

The CVCS malfunction is analyzed by employing the detailed digital computer program LOFTRAN[1]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the ECCS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

Four cases were analyzed:

1. Minimum reactivity feedback, with automatic pressurizer spray

2. Minimum reactivity feedback, without automatic pressurizer spray

3. Maximum reactivity feedback, with automatic pressurizer spray

4. Maximum reactivity feedback without automatic pressurizer spray.

The assumptions incorporated in the analyses were as follows:

1. Initial Operating Conditions

Pressurizer pressure is assumed to be initially at its minimum value. Pressurizer water level is assumed to be at the high end of the range of the values consistent with its programmed level. The initial reactor power and RCS temperature are at their full power values with uncertainties.

2. Reactivity Coefficients

   (a) Minimum reactivity feedback case

   A least negative moderator temperature coefficient and a least negative Doppler only power coefficient.

   (b) Maximum reactivity feedback case

   The most negative moderator temperature coefficient and a most negative Doppler coefficient.

3. Reactor Control

   The reactor was assumed to be in manual control. As shown in Figures 15.5-2 through 15.5-4, core power and average RCS temperature change very little. Thus, the effects of automatic rod control would be negligible.

4. Charging System

   Maximum charging system flow based on RCS back pressure from one centrifugal pump is delivered to the RCS.

5. Reactor Trips

   The transient is initiated by the pressurizer level channel which is used for control purpose failing low. As a worst single failure, another pressurizer level channel fails low, defeating the two out of three high pressurizer level trip. No reactor trips are used.

c. <u>Results</u>

Figures 15.5-2 through 15.5-4 show the transient response due to the charging system malfunction. In all the cases analyzed, core power and RCS average temperature remain relatively unchanged. Cases where the pressurizer spray is inoperable, show the pressurizer level increases at a relatively constant rate. This is because the pressurized pressure initially rises very quickly to the pressure at which the relief valves open and remains there. Cases where the pressurizer spray is operable, show the pressurizer level increases with varying rates. Spray actuation tends to keep the pressurizer pressure lower for several minutes which allows the charging pumps to deliver more flow. Eventually, pressurizer pressure does increase enough to open the relief valves.

The calculated sequence of events is shown in Table 15.5-2.

d. Conclusions

The results show that none of the operating conditions during the transient approach core limits. Because the high pressurizer level trip has been defeated by failures, the transient must be terminated by the operator. The sequence of events presented in Table 15.5-2 shows that the operator has sufficient time to take corrective action.

15.5.3 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April, 1984.

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCREASE IN REACTOR COOLANT INVENTORY EVENTS

Accident	Event	Time (seconds)
Inadvertent Actuation of ECCS During Power Operation	Spurious SI signal generated; two charging pumps begin injecting borated water	0.0
	Pressurizer fills,	505
	PORV open at 2400 psia to prevent flow through Pressurizer safety valves, or	549
	ECCS terminated without credit for PORVs to prevent flow through Pressurizer safety valves	606

TABLE 15.5-2

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
CVCS Malfunction		
1) Minimum feedback; with pressurizer spray	Two pressurizer level channels fail low; maximum charging is begun; letdown is isolated; low pressurizer level alarm	0.0
	High pressurizer level alarm	548
	Pressurizer fills	821
2) Minimum feedback; without pressurizer spray	Two pressurizer level channels fail low; maximum charging is begun; letdown is isolated; low pressurizer level alarm	0.0
	High pressurizer pressure alarm	119
	Pressurizer relief valve setpoint reached	142
	High pressurizer level alarm	796
	Pressurizer fills	1099
3) Maximum feedback; with pressurizer spray	Two pressurizer level channels fail low; maximum charging is begun; letdown is isolated; low pressurizer level alarm	0.0
	Pressurizer relief valve setpoint reached	455
	Pressurizer fills	709
4) Maximum feedback; without pressurizer spray	Two pressurizer level channels fail low; maximum charging is begun; letdown is isolated; low pressurizer level alarm	0.0
	High pressurizer pressure alarm	59
	Pressurizer relief valve setpoint reached	69
	High pressurizer level alarm	642
	Pressurizer fills	933

TABLE 15.5-3

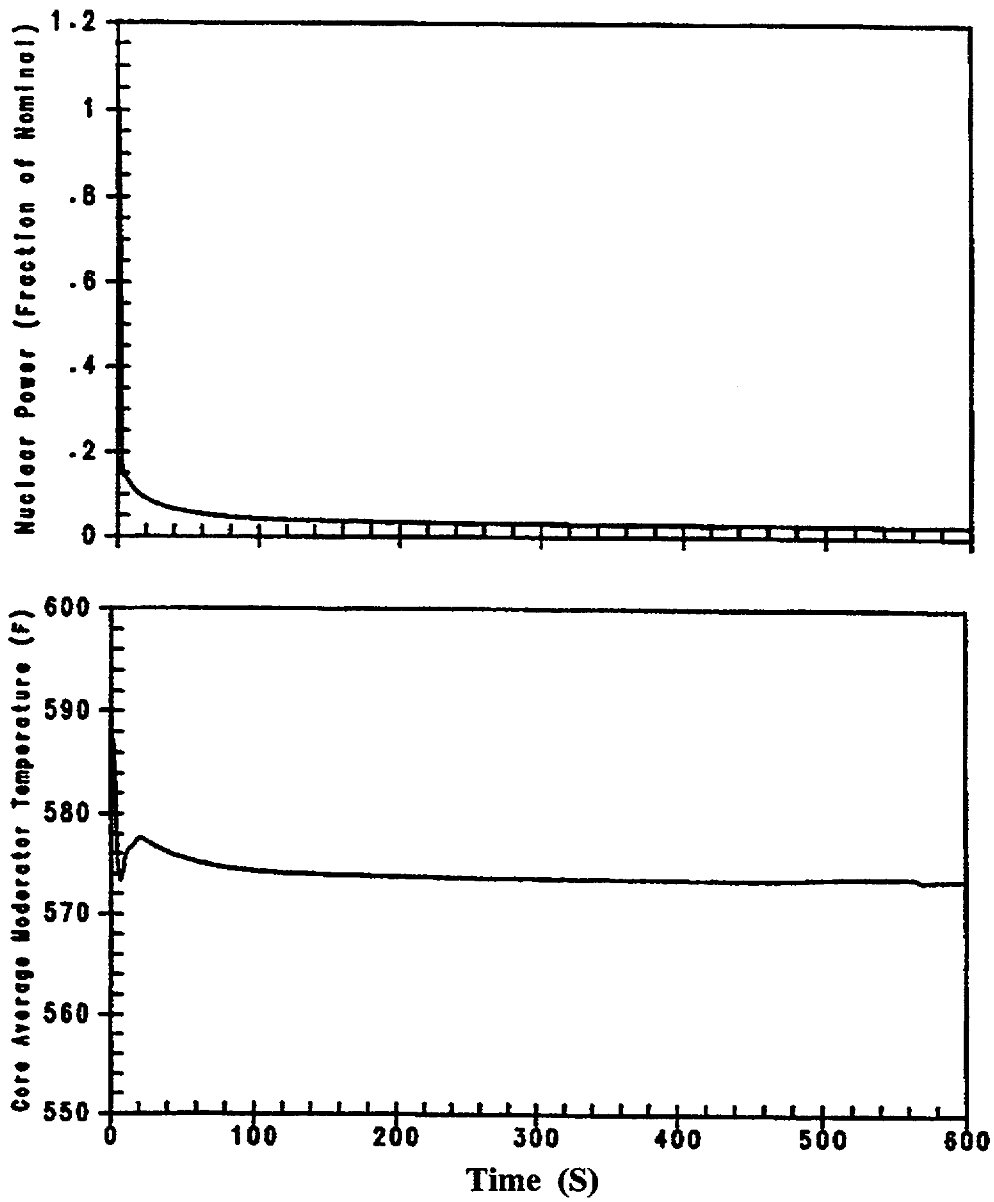
DELETED

TABLE 15.5-4

DELETED

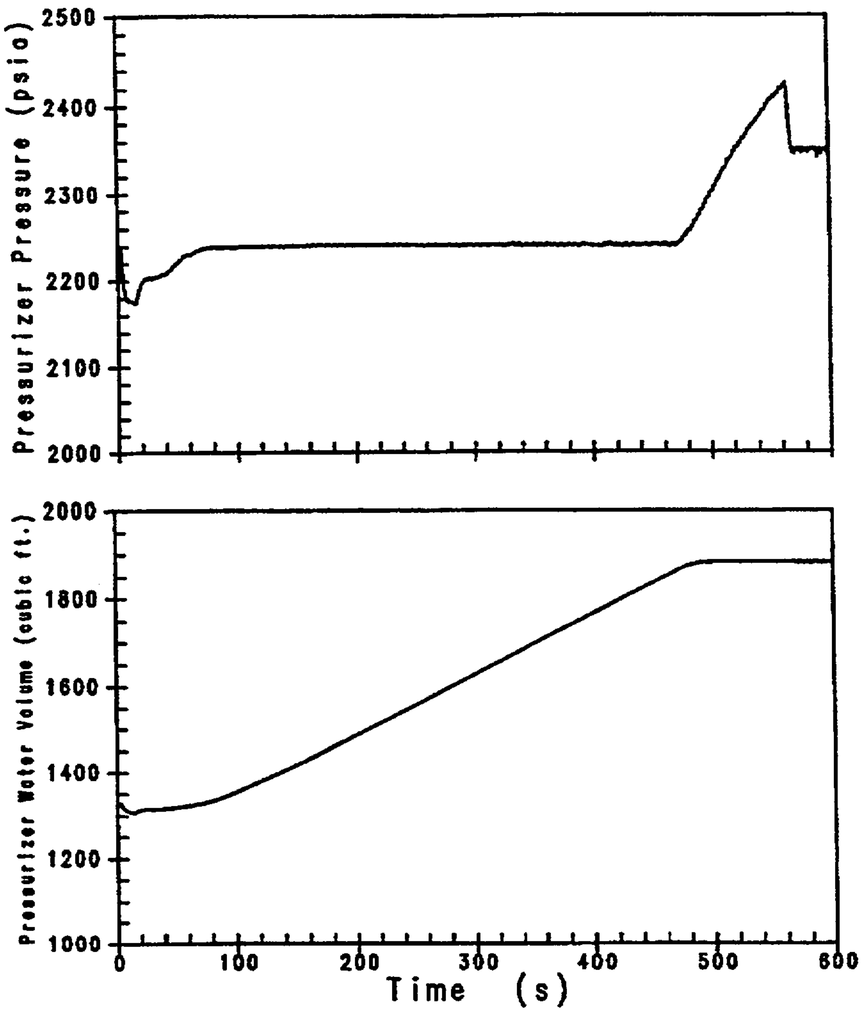
TABLE 15.5-5

DELETED



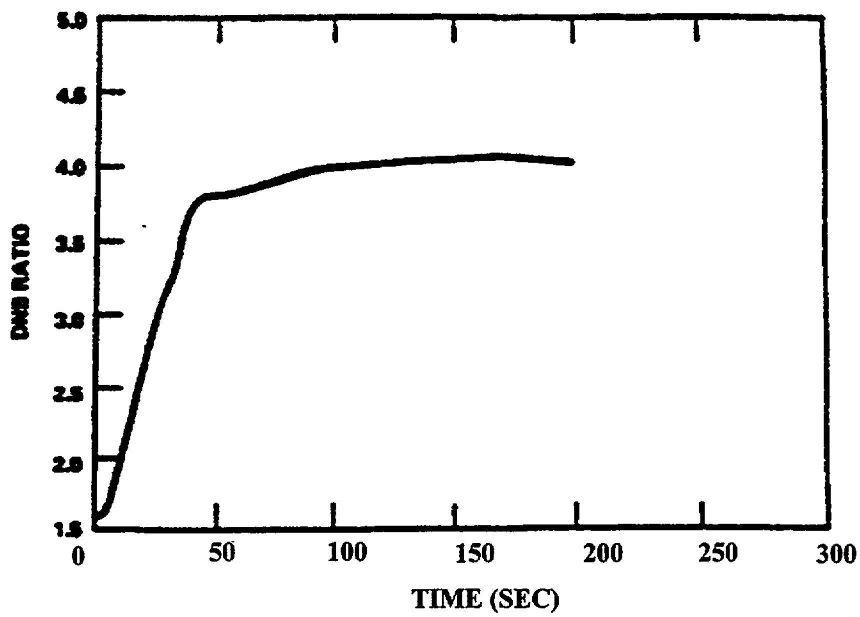
G:\Word\Images_P\Images_P.UFSAR\51211.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Coolant Temperature Transients for an Inadvertent ECCS Actuation During Power Operation	
	REV. 07	FIGURE 15.5-1 SH. 1



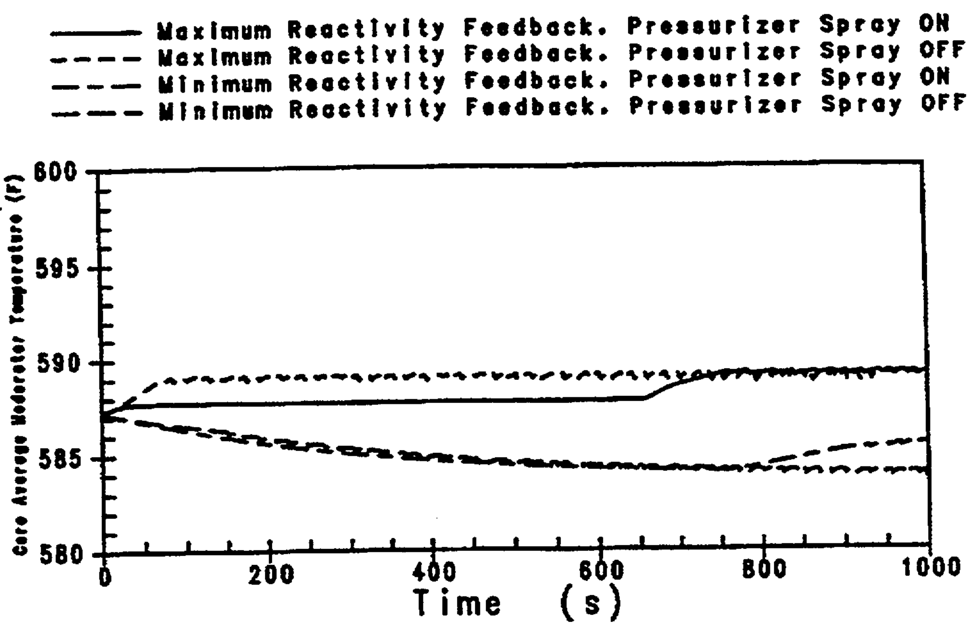
G:\Word\Images_P\Images_P.UFSAR\51212.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an inadvertent ECCS Actuation During Power Operation	
	REV. 07	FIGURE 15.5-2 SH.2



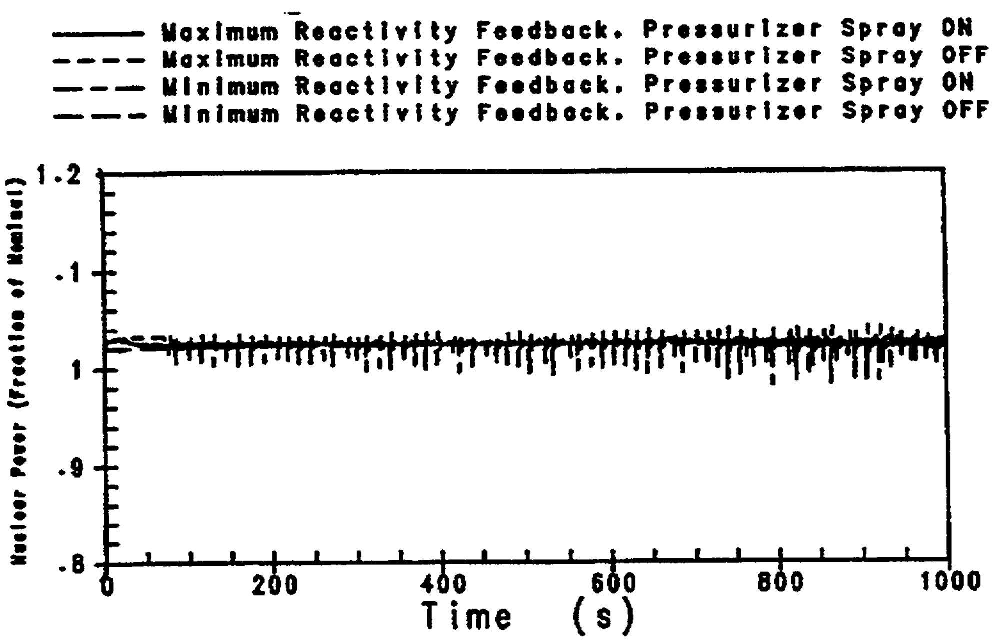
G:\Word\Images_P\Images_P.UFSAR\51213.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Flow and Typical DNBR Transients for an Inadvertent ECCS Actuation During Power Operation	
	REV. 07	FIGURE 15.5-1 SH. 3



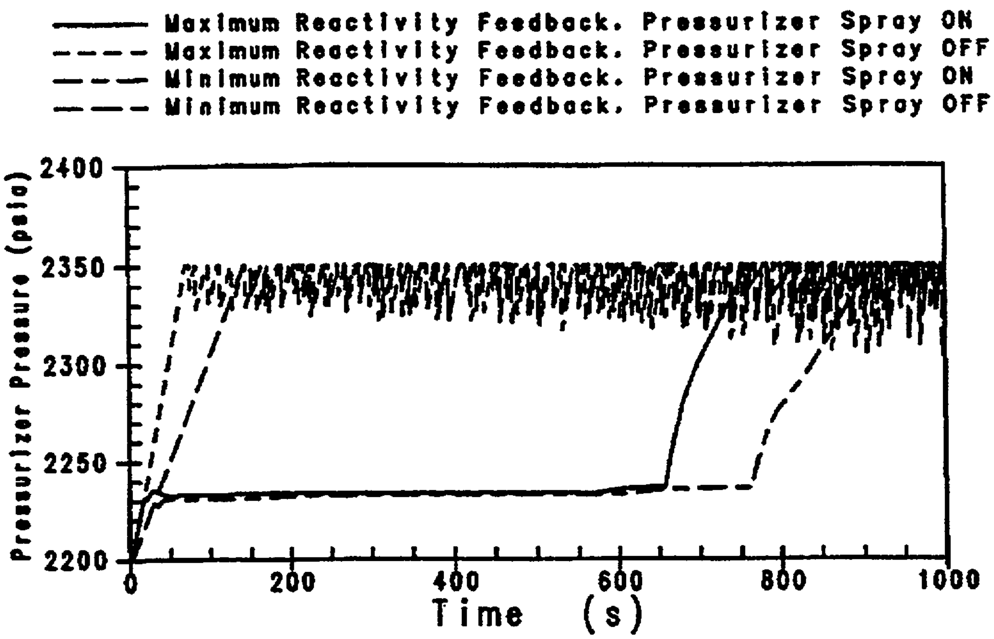
G:\Word\Images_P\Images_P.UFSAR\51221.ds4

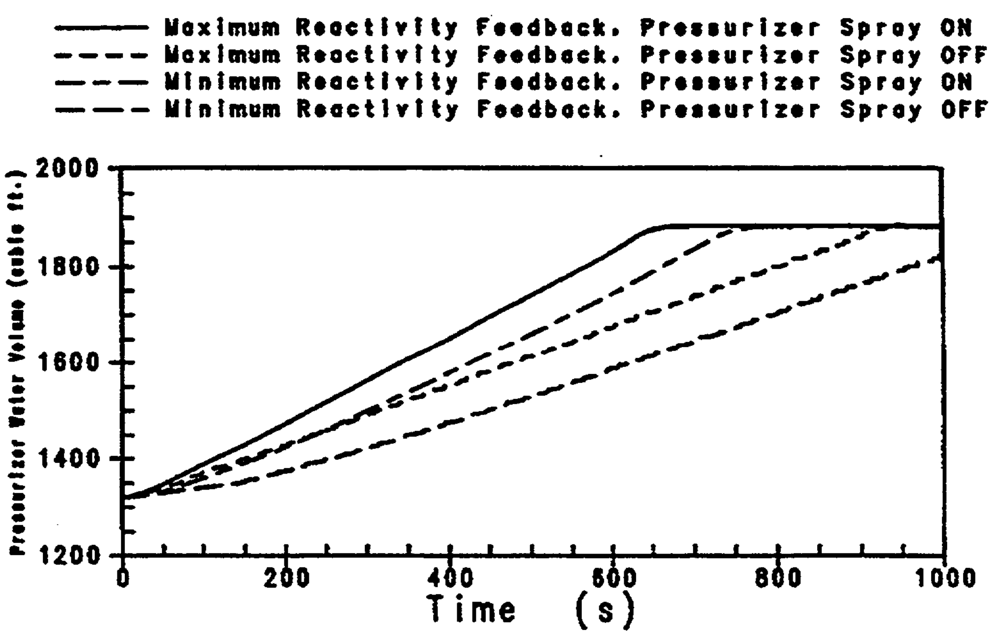
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature Transients for a CVCS Malfunction	
	REV. 07	FIGURE 15.5-2



G:\Word\Images_P\Images_P.UFSAR\51222.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power Transients for a CVCS Malfunction	
	REV. 07	FIGURE 15.5-3





G:\Word\Images_P\Images_P.UFSAR\51223.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure Transients for a CVCS Malfunction	
	REV. 07	FIGURE 15.5-4

FIGURE 15.5-5

DELETED

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

a. Inadvertent opening of a pressurizer safety or relief valve

b. Break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate Containment

c. Steam generator tube failure

d. Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback (positive MTC), but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following reactor protection system signals:

a. Overtemperature ΔT

b. Pressurizer low pressure.

An inadvertent opening of a pressurizer safety or relief valve is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN (Reference 37). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the revised thermal design procedure described in WCAP-11397[36].

In order to give conservative results in calculating the Departure from Nucleate Boiling Ratio (DNBR) during the transient, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 36.

2. The most positive MTC is assumed.

3. The least negative Doppler coefficient of reactivity is assumed such to maximize power increase prior to the reactor trip.

4. The pressurizer safety valve flowrate is assumed to be 120% of the design capacity of the valve.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the manual mode in order to prevent rod insertion due to an increase in RCS temperature prior to reactor trip. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

b. Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6-1 and 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly from the initial value until reactor trip occurs on overtemperature ΔT. The pressure transient and average coolant temperature transient following the accident are given in Figure 15.6-2. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6-1.

15.6.1.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.6.1.4 Conclusions

The results of the analysis show that the Overtemperature ΔT reactor protection system signal provides adequate protection against the RCS depressurization event. No fuel or clad damage is predicted for this accident.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

The sample lines from the hot legs of reactor coolant loops 1 and 4, and from the steam and liquid space of the pressurizer, and the chemical and volume control system (CVCS) letdown and excess letdown lines penetrate the Containment. The sample lines are provided with normally closed isolation valves on both sides of the containment wall, as sampling requirements dictate only intermittent daily use. The CVCS letdown line is provided with normally open containment isolation valves on both sides of the containment wall that are designed in accordance with the requirements of General Design Criteria 55. The excess letdown line is normally isolated and is also provided with two normally open containment isolation valves. The temperature of this fluid leaving the Containment is a maximum of 380°F.

The most severe pipe rupture with regard to radioactivity release during normal power operation is a complete severance of the 3-inch letdown line outside the Containment between the outboard containment isolation valve and the letdown heat exchanger (see Figures 9.3-26 through 9.3-29 and 9.3-31).

15.6.2.2 Analysis of Effects and Consequences

The occurrence of a complete severance of the letdown line would result in a loss-of-reactor coolant at the rate of about 140 gpm (referenced to a density of 62 lb/ft^3). This release rate is within the normal charging system makeup capability and would not result in actuation of any Engineered Safety Features Systems. Area radiation and leakage detection instrumentation provide the primary means for detection of a letdown line rupture (see Subsection 5.2.5). Frequent operation of the CVCS Reactor Makeup Control System and the other CVCS instrumentation will aid the operator in diagnosing a letdown line rupture.

The time required for the operator to identify the accident and isolate the rupture is expected to be within 30 minutes of the rupture. Once the rupture is identified, the operator would isolate the letdown line rupture by closing the high pressure letdown valves, followed by closing the pressurizer low level isolation valves. Alternatively, the operator could close the letdown line containment isolation valves to isolate the rupture. All valves are provided with control switches at the main control board. There are no single failures that would prevent isolation of the letdown line rupture.

15.6.2.3 Radiological Consequences

A conservative analysis and a realistic analysis are considered. The conservative analysis employs more pessimistic assumptions regarding fission product release and transport into the environment. The assumptions and parameters of the analyses are presented in Table 15.6-2. Detailed assumptions which are not stated in Table 15.6-2 are discussed in this Subsection.

For the conservative analysis, the activity concentration of primary coolant is assumed to be the Technical Specification limit with coincident iodine spike (see Appendix 15B). For the realistic analysis, the activity concentration of primary coolant is assumed to be the equilibrium concentration at 0.12 percent failed fuel (Table 11.1-1). Table 15.6-3 shows the primary coolant activity concentration for the conservative and realistic analyses.

It is assumed that the flow rate from the complete severance of a 3" letdown line is 140 gpm, and the time period of release is 30 minutes. The total mass of spilled reactor coolant is 1.59×10^7 gm.

It is further assumed that all the iodine in the spilled reactor coolant which flashes to steam becomes airborne and available for release to the atmosphere. The steam flashing factor for iodine is calculated to be 33 percent. All the noble gas in the spilled reactor coolant is assumed to be released to the atmosphere.

Table 15.6-4 shows the activity released to the atmosphere for the conservative and realistic analyses. The doses resulting from the letdown line break for the conservative and realistic analyses are shown in Table 15.6-5.

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary (see Section 5.2). For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Section 15.0.1).

Ruptures of small cross-section will cause expulsion of the coolant at a rate which can be accommodated by the high head safety injection pumps and which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to containment contains the fission products present in it.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an American Nuclear Society (ANS) Condition III event, which is a fault which may occur very infrequently during the life of the plant.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

a. The calculated peak fuel element clad temperature does not exceed the requirement of 2200°F.

b. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen, does not exceed one percent of the total amount of Zircaloy (or Zirlo) in the fuel rod cladding.

c. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide a significant margin in Emergency Core Cooling System (ECCS) performance following a LOCA. Reference 2 presents a study in regards to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

15.6.5.2 Sequence of Events and Systems Operations

a. Large Break LOCA Sequence of Events and Systems Operations

The Large Break LOCA (LBLOCA) Analysis of Record (AOR) for Seabrook Unit 1[29], is based on the BASH Evaluation Model (EM) [34]. The analysis therein is applicable for V5H (w/o IFMs) fuel with ZIRLO™ cladding. The LBLOCA is limited by a break discharge coefficient (C_d) = 0.6, Minimum Safeguards Injections (SI) and Beginning-of-Life (BOL) fuel conditions. The resultant AOR Peak Clad Temperature (PCT) was 1889°F for IFBA ZIRLO™ fuel.

The new fuel scheduled to be included in Seabrook Cycle 8 is similar to the V5H (w/o IFMs) fuel that is currently being used, the only significant difference being the addition of Intermediate Flow Mixer grids. The effect of this fuel feature is considerable to LBLOCA due to the importance of grids in the fuel cladding heatup calculation in the LOCBART code. In the grid model used in LOCBART, there is no heat generated in the grid, which makes grid rewet a possibility. The effect of grid rewet is increased heat transfer at the grid elevation and therefore, a decrease in the temperature of the vapor traveling through the rest of the core. This phenomenon results in a substantial PCT benefit.

The LBLOCA evaluation for the new V5H (w/ IFMs) fuel was performed using the latest version of the BASH Evaluation Model. The limiting time in life (BOL), break size (C_d = 0.6) and SI parameters (minimum) from the AOR were used as the base case conditions for this evaluation. Other important analysis assumptions, consistent with the AOR, are as follows: licensed core power of 3411 MWt, 13% uniform SGTP (8% plugging, 5% LOCA and Seismic tube crush), maximum peaking factor $F_Q(Z)$ envelope of 2.50, a hot channel enthalpy rise factor $F^N_{\Delta H}$ of 1.65, T_{COLD} = 557.6°F, T_{HOT} = 619.4°F, and Thermal Design Flow of 38,788 lbm/s.

Cosine and skewed core axial power shapes were explicitly analyzed. For non-IFBA fuel, the limiting case was an 8.5' skewed power shape case with a calculated PCT of 1786°F. The limiting IFBA case was also an 8.5' skewed power shape case with a calculated PCT of 1838°F. Burned V5H (w/o IFMs) was also analyzed with the assumption of 12,000 MWD/MTU minimum burnup. This minimum burnup will be tracked as part of the reload process. This burned V5H (w/o IFMs) was also limited by an 8.5' skewed shape and resulted in a calculated PCT of 1827°F.

The new fuel PCT calculations used current methodology and the latest version of the BASH EM computer codes such that all of the previously reported BASH EM PCT model assessments were considered. Consequently, starting with Cycle 8, the previously reported BASH EM PCT model assessments (+73°F) will no longer be part of the 10 CFR

50.46 PCT licensing basis. The assessments related to plant safety evaluations will remain. Per Reference 35, due to hydraulic differences in fuel types, a generic +50°F transition core penalty (TCP) will be applied to the V5H (w/ IFMs) fuel. This TCP will be removed upon the implementation of a full core of V5H (w/ IFMs) fuel.

This LBLOCA evaluation concludes that, with the new fuel, Seabrook remains in compliance with the requirements of 10CFR50.46. The calculated LBLOCA PCT benefit for the new fuel is 51°F (1889°F-1838°F). The net LBLOCA PCT benefit during the transition from the current fuel to the new fuel is 74°F (51°F+73°F-50°F). The net LBLOCA PCT benefit with a full core of new fuel is 124°F (51°F+73°F=124°F).

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Loss-Of-Offsite Power (LOOP) is assumed coincident with the occurrence of the break. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

- Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

For large break LOCAs, the most limiting single failure with respect to peak clad temperature (PCT) has been shown by experience to be that which reduces safety injection while producing the lowest containment pressure. The lowest containment pressure would be obtained only if all containment spray pumps operated subsequent to the postulated LOCA. Therefore, for the purposes of large break LOCA analyses, the most limiting single failure would be the loss of one residual heat removal (RHR) pump with full operation of the spray pumps. Credit could be taken for two safety injection pumps (SIPs), two centrifugal charging pumps (CCPs) and one RHR (low head) pump for a large break. However, the Seabrook large break LOCA analysis conservatively assumes both maximum containment safeguards (lowest containment pressure) and minimum ECCS safeguards (the loss of one complete train of ECCS components which includes one RHR pump, one SIP and one CCP), which results in the minimum delivered ECCS flow available to the RCS. Both Emergency Diesel Generators (EDGs) are assumed to start in the modeling of the containment spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 and is conservative for the large break LOCA. This assumption is consistent with the current methodology for large break analyses.

b. Small Break LOCA Sequence of Events and Systems Operations

The Seabrook Small Break Loss-of-Coolant Accident (SBLOCA) analysis of record (AOR) was performed to support the V5H (w/o IFMs) fuel upgrade[29]. Pertinent analysis assumptions include: licensed core power of 3411 MWt, 13% uniform SGTP, maximum peaking factor $F_Q(Z)$ envelope of 2.50, and a hot channel enthalpy rise factor $F_{\Delta H}$ of 1.65. A break spectrum of 3 inch, 4 inch, and 6 inch breaks was analyzed, resulting in the 4 inch case being limiting with a peak cladding temperature (PCT) of 1082°F [29].

It was expected that the effect of adding IFMs on SBLOCA is small as shown in past assessments for other plants. An evaluation was performed to estimate the effect of the increased fuel assembly pressure drop caused by the addition of IFMs for Seabrook. Note that IFM grids are not explicitly modeled in SBLOCA.

To determine the effect of adding IFMs, the Seabrook AOR limiting break size (4 inch) was re-run using the most recent code versions and the plant specific inputs and modeling assumptions from the AOR, which includes No SI in the Broken Loop and setting the COSI condensation model off. Subsequent to this, the 4 inch break was run with the modified THRIVE data to incorporate the revised core pressure drops associated with the new fuel. The 4 inch case re-run PCT was 1191°F. The 4 inch break with IFMs resulted in a PCT of 1186°F. This is a PCT benefit of 5°F (1191°F - 1186°F).

In summary, the Small Break LOCA evaluation for Seabrook, utilizing the currently approved NOTRUMP Evaluation Model[33], resulted in a peak clad temperature benefit of 5°F for the 4 inch diameter cold leg break. Thus, a core of V5H (w/o IFMs) bounds a core of V5H (w/ IFMs) fuel.

The results of this Small Break ECCS evaluation have shown that Seabrook remains in compliance with the requirements of 10 CFR 50.46.

Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothetical break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a function of the assumed break opening time, break area and RCS operating conditions of power, temperature and pressure. Thermal/hydraulic data which bounds the V5H (w/ IFMs) was used in generating vessel LOCA forcing functions for Seabrook Unit 1. Operating conditions were based on a Core Power of 3429.4 MWt, a Vessel Average Temperature of 588.5°F, 8% Steam Generator Tube Plugging and an RCS pressure of 2250 psia. The LOCA forces analysis also bounded a pressurizer pressure uncertainty of ±50 psi, and a temperature uncertainty of ±10°F. Leak-Before-Break methodology was used to allow consideration of branch line breaks. Therefore, the forcing functions generated were based on breaks of the accumulator line and the pressurizer surge line, which have smaller areas than the postulated breaks in the main RCS loop piping. The LOCA hydraulic forces generated for use in the reactor vessel and internals analysis bound the use of V5H (w/ IFMs) at Seabrook Unit 1.

Forces acting on the RCS loop piping as a result of the hypothesized LOCA are not significantly influenced by changes in fuel assembly design. Thus, the use of V5H (w/ IFMs) at Seabrook Unit 1 will not result in an increase of the calculated consequences of a hypothesized LOCA on the RCS piping. The current FSAR analysis for forces on RCS piping resulting from a hypothesized LOCA are considered to be bounding to the application of V5H (w/ IFMs) fuel at Seabrook Unit 1.

Post LOCA Long-Term Cooling, Subcriticality Evaluation

The Westinghouse licensing position for satisfying the requirements of 10 CFR Part 50.46 (b)(5) "Long-Term Cooling" is defined in WCAP-8339-NP-A[30], WCAP-8472-NP-A[31], and Technical Bulletin NSID-TB-86-08[32]. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out.

The primary factors affecting the calculations for the Post-LOCA long term core cooling subcriticality requirement in the Westinghouse methodologies are the RCS, RWST and accumulator water volumes and boron concentrations. The implementation of IFMs will have no affect on the volumes and boron concentrations of the RWST and accumulators used in the calculation. Therefore, implementation of IFMs will not adversely affect the assumptions made when verifying Post-LOCA long term core subcriticality for the Westinghouse ECCS Evaluation Model. The conclusions made regarding Post-LOCA subcriticality and long term cooling would remain unchanged due to implementation of IFMs for the Westinghouse ECCS Evaluation Model..

Hot Leg Switchover to Prevent Potential Boron Precipitation

The Post-LOCA Hot Leg Switchover time is determined for inclusion in the emergency procedures to preclude boron precipitation in the reactor vessel due to boiling in the core and to ensure that the core remains amenable to cooling in the long term. The Hot Leg

Switchover time is dependent on power level and the RCS, Refueling Water Storage Tank (RWST) and accumulator water volumes and boron concentrations. The implementation of IFMs will have no affect on the power level and volumes and boron concentrations of the RWST and accumulators. The RCS volume will decrease by an insignificant amount due to the additional grids. Therefore, there would be no adverse effect on the Post-LOCA Hot Leg Switchover time for implementation of IFMs for the Westinghouse ECCS Evaluation Model.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-low-pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

- Reactor trip and borated water injection, together with void formation, cause a rapid reduction of nuclear power to a residual level corresponding to delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position.

- Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

For small break LOCAs, the most limiting single active failure is the one that results in the minimum emergency core cooling system (ECCS) flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit can be taken for only one centrifugal charging pump (CCP), one safety injection pump (SIP), and one residual heat removal (RHR) (or low head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one for each loop) with one branch injection line (RHR and SIP) spilling to the RCS backpressure. Because the CCP branch injection line diameter where the break occurs may be less than the break size for small breaks, the CCP injection line is assumed to spill to containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, the SIP and CCP performance curves were degraded by 5%, the RHR pump performance curve was degraded 8.75%, and a 10 gpm flow imbalance was assumed for the high head safety injection pumps (CCP and SIP).

1. Description of Small Break LOCA Transient

The sequence of events following a small break LOCA are presented in Table 15.6-1.

Before the break occurs the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer signal. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as the pumps coast down following LOOP. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. Continued heat addition to the secondary system results in increased secondary system pressure which leads to steam relief via the main steam safety valves. Makeup to the secondary is automatically provided by the emergency feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation, control, and bypass valves and initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, for most small breaks the vessel mixture level starts to increase, covering the fuel with ECCS pumped injection before accumulator injection begins.

15.6.5.3 Core and System Performance

a. Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Reference 1).

1. Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including

f_c = coalescence factor

For Seabrook the design parameters are:

h = 129 ft

F = 2,840 gpm = 2.278×10^4 ft^3/hr

V_s = 2.310×10^6 ft^3

d = 1,000 microns = 0.100 cm

Vd = 5.42 cm/sec (based on H, iodine partition factor = 5,000)

U = 335.3 cm/sec = 11.0 ft/sec

f_m = 1 (based on using two region spray model)

f_c = 0.91

Using the "DL-1" model and the above parameters, the elemental iodine removal rate constant is calculated to be 34.3 hr^{-1}.

The conservative spray removal analysis assumes, however, a maximum removal rate of 10 hr^{-1} since credit is taken for 50 percent iodine plateout within the Containment prior to release to the environment via containment leakage. No further removal of elemental iodine is assumed when the overall containment concentration reaches 1 percent of its initial value. This concentration is reached in approximately 0.83 hours after spray actuation. The iodine removal function and effectiveness of the Containment Spray System is discussed in Subsections 6.5.2 and 6.5.3. Credit for iodine removal by the spray system is assumed to start 62 seconds after onset of the LOCA.

The removal rate of particulate iodine by the spray system is also based on the WASH-1329 model and is calculated as follows:

$$\lambda p = \frac{3 \, h \, Ep \, F}{2 \, d \, V_s} f_c$$

Where:

λp = particulate iodine removal rate constant, hr^{-1}

h = spray fall height, ft

E_p = total collection efficiency of particulate iodines of a single drop

F = spray flow rate, ft^3/hr

d = drop diameter, ft

V_s = containment sprayed volume, ft^3

f_c = coalescence factor

For Seabrook the parameters used are as follows:

h = 129 ft

E_p = $1.5x10^{-3}$

F = $2.278x10^4$ ft^3/hr

d = $3.28x10^{-3}$ ft (1,000)

V_s = $2.310x10^6$ ft^3

f_c = 0.91

Thus, λp is calculated to be 0.79 hr^{-1}. The conservative analysis assumes, however, a maximum particulate iodine removal rate of 0.45 hr^{-1} as suggested in WASH-1329. The time for the containment concentration of particulate iodines to reach 1 percent of its initial value is calculated to be 12.1 hours. No removal of organic iodine by the spray system is assumed.

2. Realistic Analysis

The spray removal constants for both elemental and particulate iodine in the realistic analysis are represented by the same equations and parameters as in the conservative case, with the following exceptions. No upper limits to the iodine spray removal constants are assumed and the spray system is assumed to be effective for iodine removal for the time it takes to decrease initial iodine concentrations to 0.5 percent. The resultant spray removal rate for elemental iodines is 36.2 hr^{-1}, applied for 0.37 hours, and 0.84 hr^{-1} for particulate iodines, applied for a 5.7 hour period.

No removal of organic iodine by the spray system is assumed and no credit for the spray system is taken for the first 62 seconds following a LOCA.

9. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-proprietary), April 1975.

10. Special Report NS-NRC-85-3025(NP), "BART-WREFLOOD Input Revision."

11. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

12. Collier, G., et al, "BART-Al: A Computer Code for the Best Estimate Analysis of Reflood Transients" WCAP-9561, January 1980.

13. "Westinghouse ECCS - Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-proprietary), July 1974.

14. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-proprietary), July 1974.

15. Johnson, W. J.; Massie, H. W.; and Thompson, C. M. "Westinghouse ECCS Four Loop Plant (17x17) Sensitivity Studies,5 WCAP-8586-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

16. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code,5 WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

17. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and 10081-A (Nonproprietary), August 1985.

18. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants," NUREG-0611, January 1980.

19. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8301 (Proprietary), June 1974.

20. "Clarification of TMI Action Plan Requirements", NUREG-0737, November 1980.

21. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31", November 2, 1983.

22. Rupprecht, S. D. et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary), and WCAP-11372-A (Non-Proprietary), October 1986.

23. U.S. Nuclear Regulatory Commission (USNRC) Standard Review Plan (SRP) 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," NUREG-0800, Revision 2, July 1981.

24. EPRI NP-1850, Volume 1, Rev. 4, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics (Rev. 4)", Electric Power Research Institute, November 1988.

25. NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," A. K. Postma and S. Tam, January 1978.

26. EPRI-4787, "Tests of Steam Generator Transient Response to Scenarios Involving Steam Generator Tube Ruptures and Stuck-Open Safety Relief Valves," K. Garbett, O. J. Mendler, G. C. Gardner, R. Garnsey, and M. Y. Young, May 1987.

27. GENRUP, "A Computer Code for the Radiological Assessment of Steam Generator Tube Rupture Accidents," J. N. Hamawi, October 1987.

28. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995, (Westinghouse Proprietary)

29. 93NA-G-0037, "North Atlantic Energy Service Corporation Seabrook Station LOCA Safety Analysis Report," August 31, 1993.

30. WCAP-8339-NP-A, "Westinghouse ECCAS Evaluation Model Summary," Bordelon, F. M., July 1974.

31. WCAP-8471-P-A, WCAP-8472-A, "Westinghouse ECCS Evaluation Model: Supplementary Information," Bordelon, F. M., April 1975.

32. NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," October 31, 1986.

33. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," Lee, N., et al., August 1985.

34. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", May 1988.

35. WCAP-10444-P-A, "VANTAGE 5H Fuel Assembly", April 1988.

36. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984.

37. WCAP-7907-P-A, "LOFTRAN Code Description," T.W.T. Burnett, et al., April 1984.

TABLE 15.6-1
(Sheet 1 of 4)

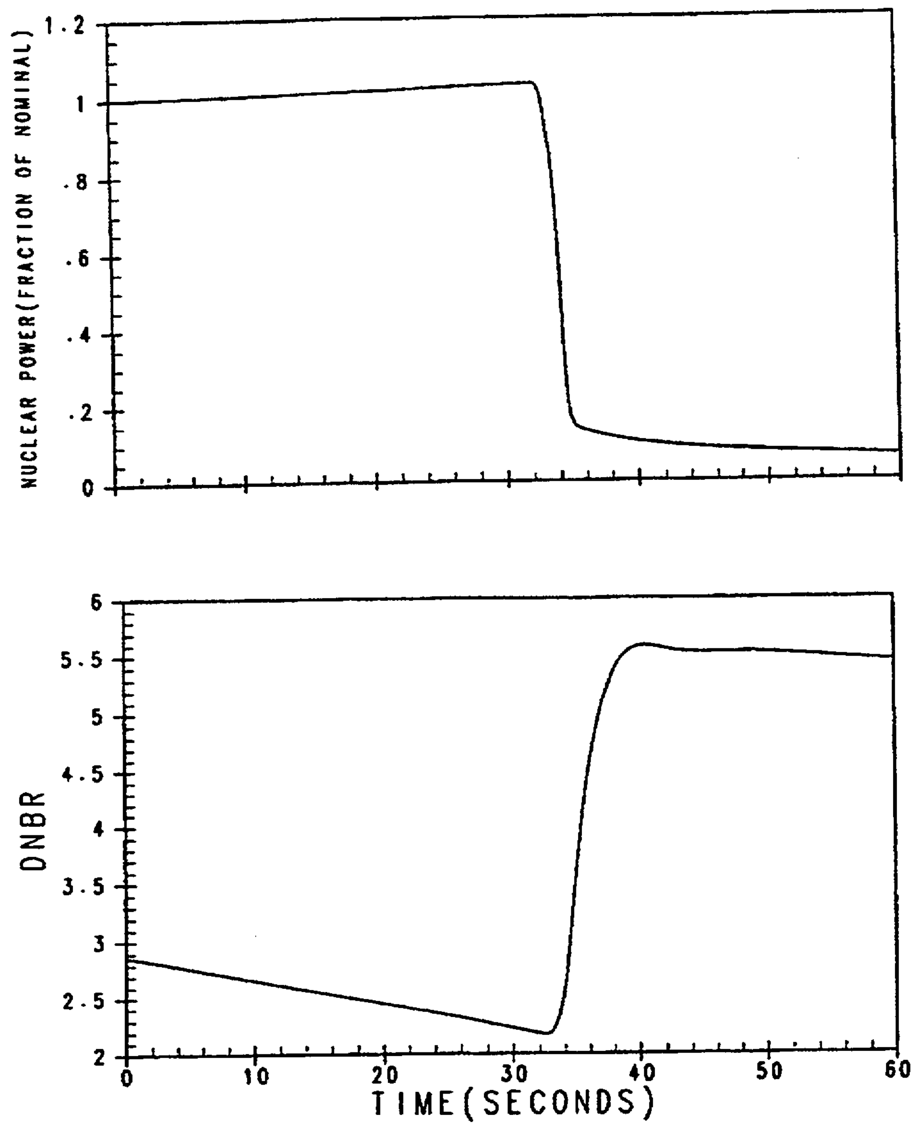
TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (Seconds)
a. Inadvertent Opening of a Pressurizer Safety Valve	Safety valve fully opens	0.0
	Overtemperature ΔT reactor trip setpoint reached	29.5
	Rods begin to fall into core	32.0
	Minimum DNBR occurs	32.6
b. Large Break LOCA		
1. DECLG $C_D = 0.8$ Minimum SI	Start	0
	Reactor trip signal	0.72
	Safety injection signal	1.45
	Accumulator injection begins	12.5
	End-of-bypass	28.9
	End-of-Blowdown	28.9
	Pump injection begins	31.5
	Bottom of core recovery	43.8
	Accumulator empty	50.1
2. DECLG $C_D = 0.6$ Minimum SI	Start	0
	Reactor trip signal	0.735
	Safety injection signal	1.67

TABLE 15.6-1
(Sheet 2 of 4)

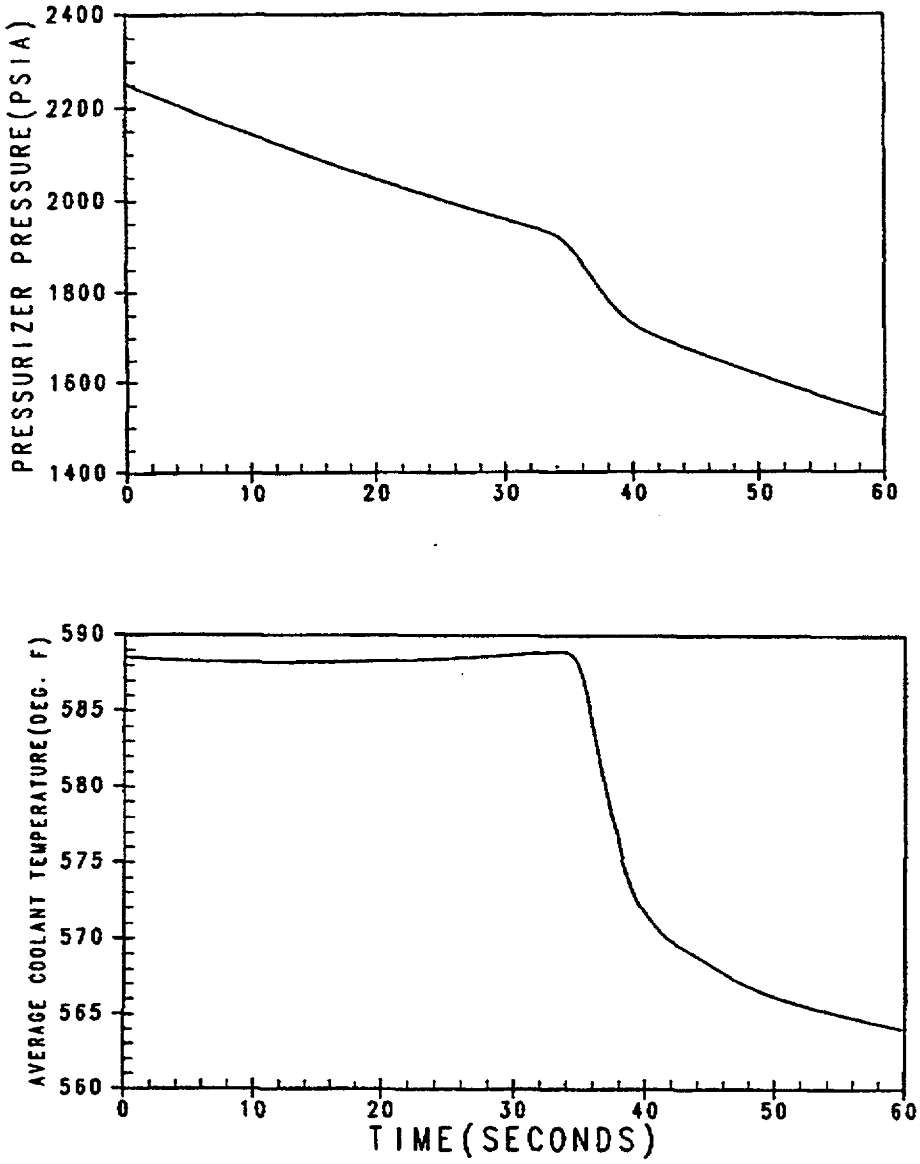
TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident		Event	Time (Seconds)
		Accumulator injection begins	15.0
		End-of-bypass	33.0
		End-of-blowdown	33.3
		Pump injection begins	31.7
		Bottom of core recovery	47.8
		Accumulator empty	53.3
3.	DECLG C_D = 0.4 Minimum SI	Start	0
		Reactor trip signal	0.757
		Safety injection signal	2.09
		Accumulator injection begins	20.0
		Pump injection begins	32.1
		End-of-bypass	39.2
		End-of-blowdown	39.2
		Bottom of core recovery	54.5
		Accumulator empty	61.1
4.	DECLG C_D = 0.6 Maximum SI	Start	0
		Reactor trip signal	0.735
		Safety injection signal	1.67
		Accumulator injection begins	15.0
		Pump injection begins	31.7



G:\Word\Images_P\Images_P.UFSAR\51231.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and DNBR Transients for an Inadvertent Opening of a Pressurizer Safety Valve	
	REV. 07	FIGURE 15.6-1



G:\Word\Images_P\Images_P.UFSAR\51232.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Vessel Average Temperature Transients for an Inadvertent Opening of a Pressurizer Safety Valve	
	REV. 07	FIGURE 15.6-2

V_w = The average velocity of the groundwater

r = The ratio of the weight of mineral to volume of water per unit volume of aquifer material

K_d = The distribution coefficient of the given ionic species for the prevailing conditions

The quantity $(1 + K_d)$ is referred to as the retardation factor.

The distribution coefficient (K_d) assumed in this evaluation (70 ml/g) is that value used for ^{90}Sr in Reference 5. This assumption is conservative in that cesium isotopes are more tightly bound by soil than strontium isotopes and will exhibit a larger distribution coefficient. Seabrook Station and the standard site used in Reference 5 are both coastal sites with similar soil parameters and groundwater flow rates.

With the above assumptions and parameters listed in Table 15.7-16, the average velocity of Sr and Cs isotopes is calculated to be 1.2×10^{-3} ft/day. The time required to travel 200 feet to the marsh area is 457 years. Based on 457 years decay time for cesium and strontium isotopes, and 290 days decay time for all other radionuclides released, specific nuclide concentrations at the marsh are calculated and listed in Table 15.7-17. These values are based on the assumption that 80 percent of the maximum liquid volume of the affected tank is released. No credit is taken for dilution in the groundwater or liquid retention by unlined building foundations or leakage barriers.

The marsh concentration of radioisotopes is subject to tidal flushing as well as wind and wave action into Hampton Harbor. The discharge into the marsh will be quickly diluted and mixed in the intertidal zone or tidal prism of Hampton Harbor. A value of 2.24×10^8 ft^3 has been conservatively used in determining the extent of radionuclide dilution, since no credit is taken for dilution within the tidal prism of the marsh. Within the entire Hampton Harbor and estuary, the volume of the tidal prism is approximately 4.70×10^8 ft^3.

Water is lost from the entire Hampton estuary at an average rate of 9850 ft^3/sec. Expressed on a percentage basis, about 88 percent of the estuary volume leaves and returns on each ebb and flood tide cycle. At ebb slack tide, the estuarine residual is approximately 12 percent of the total volume of the basin. These figures indicate that the Hampton Harbor estuary exhibits substantial tidal exchange rate under natural conditions.

Radionuclide concentrations in Hampton Harbor can be found in Table 15.7-17, and have been used to calculate doses to individuals by the ingestion of finfish and invertebrates. Doses have been calculated based on methodology and dose conversion factors, bioaccumulation factors and maximum usage factors delineated in Regulatory Guide 1.109, Revision 1. The highest organ dose was to the lower large intestine of an adult and was calculated to be 18.2 mrem. The adult total body dose is 1.4 mrem.

15.7.3.4 Conclusions

The radioactive liquid release from a tank rupture will not result in any uncontrolled surface release. The liquid will be processed in the Liquid Waste and/or Boron Recovery System and the final effluent will be controlled and monitored before discharging into the circulating water tunnels or disposed offsite. The seepage of tank contents through cracks in the concrete cubicles will not significantly impact any potable water supply or possible ingestion pathways.

15.7.4 Fuel Handling Accident

15.7.4.1 Identification of Causes and Accident Description

Subsequent to plant start-up, a Licensing Amendment Request (LAR), LAR 94-06, "Revision to Technical Specification 3.9.4," was submitted and accepted by the staff. LAR 94-06 proposed two changes to the Seabrook Station Technical Specifications that address containment building penetrations. The first change is to allow the use of alternate closure methodologies for containment building penetrations during core alterations or movement of irradiated fuel within containment. The second change would allow both personnel airlocks to be open during core alterations or movement of irradiated fuel within containment. Consequently, the most limiting fuel handling accident is defined as the dropping of a spent fuel assembly within an open containment, resulting in the rupture of the cladding of all the fuel rods in the assembly, despite administrative controls and physical limitations imposed on fuel handling operations (see UFSAR Subsection 9.1.4). This potential fuel handling accident is considered an ANS Condition IV event, a limiting fault, since it includes the potential for significant amounts of radioactive releases. All refueling operations are conducted in accordance with prescribed procedures.

Dropping or damaging an assembly within the Fuel Storage Building (FSB) is another postulated accident addressed in this analysis. Dropping an assembly within the FSB is evaluated assuming a minimum of 23 feet of water above the assembly is available for iodine scrubbing (effective iodine decontamination factor of 100) prior to release of fuel assembly gap activity to the FSB atmosphere. Damaging an assembly (i.e., during fuel assembly maintenance or inspection) assumes a minimum of 10 feet of water above the assembly release point (effective iodine DF of 37) prior to release of fuel assembly gap activity to the FSB atmosphere.

15.7.4.2 Analysis of Effects and Consequences

a. Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident:

1. The accident occurs 100 hours following reactor shutdown, the earliest time when spent fuel would be first moved from the reactor vessel.

2. The accident results in the rupture of the cladding of all fuel rods in the assembly.

3. The damaged assembly was the one operating at the highest power level in the core region to be discharged.

   The power in this assembly, and the corresponding fuel temperatures, establish the total fission product inventory and the fraction of this inventory present in the fuel pellet-cladding gap at the time of reactor shutdown.

4. The fuel pellet-cladding gap inventory of fission products is released to the refueling cavity or spent fuel pool water at the time of the accident.

5. Refueling cavity or spent fuel pool water will retain a large fraction of the gap activity of halogens by virtue of their solubility and hydrolysis. Noble gases are not retained by the water as they are not subject to hydrolytic reactions.

Additional assumptions are given in Table 15.7-18.

b. <u>Fission Product Inventories</u>

The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. The parameters used for the calculation of the fission product inventory of the highest rated assembly to be discharged are summarized in Table 15.7-19. Tables 15.7-20 and 15.7-22 show the activity of the highest rated fuel assembly at the time of reactor shutdown and after 100 hours decay for the conservative and realistic analyses, respectively.

The conservative parameters are based on Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 23, 1972.

c. <u>Iodine Decontamination Factors</u>

An experimental test program (Reference 6) was conducted by Westinghouse to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the solution in the spent fuel pool to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid, and is controlled by the bubble diameter and contact time of the bubble in the solution.

To obtain all the necessary information regarding this mass transfer process, a number of small-scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry), and data were collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large-scale tests were also performed with carbon dioxide. The small-scale carbon dioxide tests also resulted in a mathematical expression for decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full-size fuel assembly simulator was fabricated and placed in a deep pool for testing, where gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas, and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small-scale tests with carbon dioxide, permitted an in situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large-scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small-scale iodine testing.

$$\text{Decontamination factor} = 73\, e^{0.313\, t/d}$$

where:

t = rise time

d = effective bubble diameter

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the spent fuel pool solution, and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

With consideration given to the total quantity of gas released from a full assembly, that is, 6.9 SCF for the 17x17 array, the pool decontamination factor for iodine is indicated to be a minimum of 589 for a 23 foot depth and 181 for a 10 foot depth. In the conservative analyses, a lower decontamination factor is selected to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. For the dropped assembly analysis, an overall effective iodine decontamination factor of 100 is used, as discussed in Regulatory Guide 1.25. In both the realistic and conservative evaluations, a decontamination factor of one is used for noble gas isotopes. The activity released from the spent fuel pool surface or refueling canal by isotope is shown in Tables 15.7-21 and 15.7-23 for the conservative and realistic analyses respectively.

Potential damaging of a fuel assembly in the FSB during maintenance, transfer or inspection could occur with a minimum of 10 feet of water above the assembly. As discussed above the calculated iodine DF for 10 feet of water is 181 using the analytical expression above. For the analysis described below for the potential damaging of an assembly while suspended with only 10 feet of water above the assembly, a reduced iodine DR of 37 is used to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. This value was determined by appropriate normalization of the experimental test case values.

15.7.4.3 <u>Radiological Consequences</u>

a. <u>Fuel Handling Accident Inside Containment</u>

In the event that a fuel assembly is dropped during fuel handling operations in the Reactor Containment building, operating procedures will require personnel to leave immediately, securing both open Personnel Air Locks (PALs) behind them and to isolate the Containment Purge System. Additionally, designated open containment penetrations will be ordered closed. As a backup to operator action, redundant area radiation monitors in the vicinity of the manipulator crane (see Figure 12.3-3) will alarm on high activity and automatically secure containment purge. These actions will insure that a minimum of activity will be released to the environment at the time of the accident while allowing for timely containment evacuation. Airborne and area radiation monitors are further discussed in Subsection 12.3.4 and detailed in Figure 12.3-3. Subsequently, the containment atmosphere would be analyzed to determine the airborne activity concentrations, allowing for an assessment of the extent of damage to the dropped assembly. Table 15.7-21 shows the maximum containment air concentrations (assuming no release to the environment). In the event that airborne activity is detected, the containment will be vented through particulate and charcoal filters to allow for personnel re-entry.

As a bounding estimate for potential releases prior to containment isolation, an evaluation of offsite and control room doses has been performed to confirm that doses would be well within the values specified in 10 CFR part 100, "Reactor Site Criteria," and General Design Criterion 19 for control room doses. The analysis follows the guidance of Standard Review Plan (SRP) 15.7.4 and Regulatory Guide 1.25. Specific analysis input and assumptions are discussed below. Resulting offsite (EAB) and control room doses are presented in Table 15.7-25a.

Parameters Used to Evaluate The Fuel Handling Accident Inside Containment

1. All rods contained in one fuel assembly are assumed to rupture and release the rod gap noble gas and volatile iodine fission product inventory. This source term is consistent with Regulatory Guide 1.25.

2. Fuel rod gap activities consist of 10% iodine, and 10% noble gases (except 30% Kr-85) contained in the maximum rated fuel assembly. These release fractions are consistent with Regulatory Guide 1.25 and have been confirmed for 3411 MWth operation using the bounding mechanical design power history.

3. Maximum rated fuel assembly gaseous iodine and noble gas isotopic inventories are given in Table 15.7-20. Nuclear characteristics of the highest rated discharged assembly are given in Table 15.7-19.

4. Equilibrium core fission product inventories are based on extended operation at 3654 MWt which is approximately 107% of the rated thermal power of 3411 Mwt.

5. The radiological inventory in the single damaged assembly is based on assumed operation with a 1.65 radial peaking factor, consistent with Regulatory Guide 1.25.

6. Consistent with Regulatory Guide 1.25, assembly gap iodine inventory is composed of 99.75% inorganic and 0.25% organic iodine.

7. The fuel handling accident is assumed to occur 100 hours after plant shutdown, consistent with Seabrook Station Technical Specification 3/4.9.3.

8. The fuel handling accident within the containment does not result in containment pressurization. The conservative containment release model is based on releasing all of the radioactive gases from the containment, exponentially over a two hour period. This is equivalent to releasing 77% of the available activity in 0.5 hours, 95% in 1 hour and 99.75% in two hours. The site area boundary doses are calculated over a 0-2 hour time period, while the control room doses are evaluated over a 0-30 day release period. The 0-2 hour total release rate assumption is bounding for the occurrence of both

the Sludge Lance penetrations and the PAL's being open at the same time.

9. Control room doses are evaluated for a 30 day time period using the containment release model given above and continuing for 30 days. The control room emergency filtration system is activated by the redundant safety grade intake radiation monitors, with an assumed 2 second filter bypass period. Control room intake and recirculation filter efficiencies are 95% for elemental and organic and 99% for particulate iodine species. The control room emergency filtration system operates at 2000 cfm consisting of 800 scfm of recirculated air and 1200 scfm of outside air. Outside air is from two remote 600 scfm fresh air intakes, one of which is assumed to be drawing in clean air. The control room free air volume is assumed to be 2.46E+05 ft^3. A discussion of control room habitability during a design bases event is given in UFSAR Subsection 15.6.5.4.e.

10. No credit is taken for containment recirculation filters.

11. Accident atmospheric dispersion factors assume a ground level release as given in Tables 15B-4 (for the EAB) and Table 15B-6 for the control room. Control room intake X/Q's are divided by two to reflect clean air intake by one of the two remote intakes.

12. No credit is taken for plume ground deposition or radioactive decay in transit.

13. Dose conversion factors (DCFs) for evaluating the thyroid and Effective Dose Equivalent Whole Body Doses are consistent with ICRP 30.

14. The decontamination factor for noble gases in refueling cavity water is 1.0 consistent with Regulatory Guide 1.25.

15. The total effective iodine decontamination factor is assumed to be 100, consistent with Regulatory Guide 1.25. A minimum water depth of 23 feet above the ruptured assembly is assumed.

16. Buildup of nuclide daughter products is taken into account.

17. No credit for containment iodine plateout.

18. The breathing rate given in Regulatory Guide 1.25 of 3.47E-04 m^3/sec is used for both EAB and control room doses. Table 15.7-24a lists the activity by isotope released to the environment for the 0-2 hour PAL release path release pathway. Exclusion Area Boundary (EAB) 0-2 hour resulting doses and 0-30 day control room doses for the containment fuel handling

accident are presented in Table 15.7-25a.

b. <u>Fuel Handling Accident inside Fuel Storage Building</u>

Two cases are analyzed for the potential fuel handling accident inside of the fuel storage building. Case 1 analyzes the dropping of an assembly with an assumed minimum of 23 feet of water above the release point with the overall effective iodine DF equal to 100. Case 2 analyzes the potential damaging of an assembly while in transit or suspended below a minimum of 10 feet of water above the release point with the effective overall iodine DF equal to 37. Both cases conservatively assume that the complete assembly gap inventory is released to the pool water. Additional common parameters for both of the fuel handling accidents are outlined below.

<u>Parameters Used to Evaluate the Fuel Handling Accident inside the Fuel Storage Building</u>

1. All of the rods contained in one fuel assembly are assumed to rupture and release the rod gap noble gas and volatile iodine fission product inventory. This source term is consistent with Regulatory Guide 1.25.

2. Fuel rod gap activities consist of 10% iodine, and 10% noble gases (except 30% Kr-85) contained in the maximum rated fuel assembly. This release fraction is consistent with Regulatory Guide 1.25.

3. Maximum rated fuel assembly gaseous iodine and noble gas isotopic inventories are given in Table 15.7-20. Nuclear characteristics of the highest rated discharged assembly are given in Table 15.7-19.

4. Equilibrium core fission product inventories are based on extended operation at 3654 MWt which is approximately 107% of the rated thermal power of 3411 MWt.

5. The activity inventory in the single damaged assembly is based on operation with a 1.65 radial peaking factor, consistent with R. G. 1.25.

6. Consistent with R. G. 1.25, the assembly gap iodine inventory is composed of 99.75% inorganic and 0.25% organic iodine.

7. The fuel handling accident is assumed to occur 100 hours after plant shutdown, consistent with Seabrook Technical Specification 3/4.9.3.

8. EAB accident atmospheric dispersion factors assume a ground level release as given in Table 15B-4. No credit is taken for ground deposition or decay in transit.

9. Dose conversion factors for evaluating the thyroid and Effective Dose Equivalent Whole Body doses are consistent with ICRP-30 for Case 2 and with Regulatory Guide 1.109 for Case 1. Note Case 1 is being maintained for historical purpose consistency with the original licensing basis.

10. The decontamination factor for noble gases in the refueling cavity water is one, consistent with R. G. 1.25.

11. Buildup of nuclide daughter products is taken into account.

In the event that a fuel assembly is dropped in the Fuel Storage Building, personnel must leave the building. The normal building exhaust system (Subsection 9.4.2) is isolated before initiation of fuel handling operations. The building ventilation system is in the fuel handling mode (Subsection 6.5.1) during fuel handling operations. A radiation monitor in the area will alarm on high activity. The ventilation system operates in the fuel handling mode, using the emergency air cleanup subsystem whenever irradiated fuel outside of a sealed cask is handled to maintain the building at a negative pressure of 0.25" W.G. Therefore, any activity will be diverted to the safety class filters prior to release to the environment. Operation of the Fuel Storage Building Ventilation System in the fuel handling mode is further discussed in Subsection 6.5.1, Engineered Safety Feature (ESF) Filter System.

Tables 15.7-21 and 15.7-23 show maximum fuel storage building air activity concentrations for the conservative and realistic cases respectively for dropped assembly. The airborne iodine released from the water surface consists of 75 percent inorganic iodine and 25 percent organic iodine (Regulatory Guide 1.25, 3/23/72). It is assumed all cases that all activity is released to the environment within two hours. Table 15.7-24 lists the activity by isotope which is released to the environment for the dropped assembly case. The resulting doses are shown in Table 15.7-25 for both cases.

15.7.4.4 Conclusions

The doses calculated for both the fuel handling accident occurring within the containment and for the fuel handling accident occurring in the fuel storage building are well within the values specified in 10 CFR part 100, "Reactor Site Criteria," and General Design Criterion 19 for the Control room. The fuel handling accident occurring within the containment with an open release path to the environment is the bounding fuel handling event.

15.7.5 Spent Fuel Cask Drop Accident

15.7.5.1 Identification of Causes and Accident Description

As discussed in Subsection 9.1.4, an isolation gate installed between the spent fuel pool and the transfer canal will prevent a loss of spent fuel pool water due to a postulated cask drop accident. This gate is closed during cask handling operations. The cask handling crane cannot be passed over the isolation gate or any part of the spent fuel storage area; hence, the spent fuel shipping cask cannot be transported over these areas. Consequently, in the event that a heavy cask were dropped, the spent fuel storage area integrity would not be compromised nor any stored fuel damaged. The limited travel of the cask handling crane prevents it from traveling over any safety-related equipment.

The cask is lifted in and out of the cask loading pool in two steps. The first step is from elevation (-) 23'-10½" to a shelf at elevation 4'-5¼", a lift of 28'-3¾". The second step is from elevation 4'-5¼" to clear the operating floor at the 25' elevation, a lift of 21'-6¾". The Engineered Safety Features Filter System (Subsection 6.5.1) is in operation during handling of a loaded cask. Operation of the Fuel Storage Building Ventilation System in the emergency mode is further discussed in Subsection 6.5.1.

15.7.5.2 Radiological Consequences

The radiological consequences for the postulated spent fuel cask drop accident have been calculated based on no impact limiting devices in the designs of the Seabrook Station cask handling equipment. The cask handling crane cannot be passed over the fuel pool isolation gate or any part of the spent fuel storage area; hence, the only source of fission products available for release are those contained within the spent fuel cask and the contained fuel assemblies. For the purpose of this analysis, it is conservatively assumed that all of the fuel pins are breached, releasing all of the halogens and noble gases contained in the gap area of the fuel pins.

The following assumptions are postulated in the calculation of the radiological consequences of the spent fuel cask drop accident; additional parameters are given in Table 15.7-26.

a. Conservative Analysis

    1. The maximum number of fuel assemblies contained within one shipping cask is seven assemblies, which have been stored and decayed for a minimum of 150 days. This is a conservative estimate based on methodology used in WASH 1238 "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," USAEC, December 1972, and current practices of storing spent fuel assemblies. Conservative case core (193 assemblies) and gap activities for iodines and noble gases are given in Table 15.0.6.

2. The postulated cask drop occurs within the Spent Fuel Storage Building. The Engineered Safety Features Filter System is in operation, providing an iodine DF of 20.

3. It is assumed, for the purpose of providing offsite dose consequences, that all of the fission products released within the cask are instantaneously released to the Fuel Storage Building environment and ultimately to the environment via the ESF filter system. This is very conservative in view of the stringent testing criteria (10 CFR 71, Appendix B) that spent

fuel shipping casks must comply with. The activity released as a result of the postulated accident is given in Table 15.7-27.

Offsite doses resulting from the spent fuel cask drop accident were evaluated based on the dose methodology given in Appendix 15A and the one hour atmospheric dispersion parameters presented in Appendix 15B. Results are given in Table 15.7-28.

b. Realistic Analysis

The realistic analysis assumes the realistic gap fractions (Table 15.7-22) and that the Fuel Storage Building Engineered Safety Features Filter System is 99 percent efficient for the removal of elemental iodine. Realistic doses are evaluated based on the dose methodology presented in Appendix 15A and the realistic one-hour atmospheric dispersion factor (X/Q) given in Appendix 15B. Results are given in Table 15.7-28.

15.7.6 References

1. Underhill, D.W., "Effect of Rupture in a Fission Gas Holdup Bed," Nuclear Safety, 13(6), November-December 1972.

2. S. Iwai, Y. Inove and K. Nishimaki, "Movement Through Soil of Radioactive Nuclides Contained in Chemical Processing Waste," Kyoto University, 1968.

3. "Comments on the Content of AEC's Proposed Environmental Impact Statement on the Hanford Site," US EPA, Office of Radiation Programs, 1973.

4. H.B. Levy, "On Evaluating the Hazards of Groundwater Contamination by Radioactivity from an Underground Nuclear Explosion," Lawrence Livermore Laboratory, Rept. UCRL-51278, 1972.

5. Y. Inove and S. Morisawa, "On the Selection of a Ground Disposal Site by Sensitivity Analysis," Health Phys. 26, 251-261 (1973).

6. D.D. Malinowski et al., "Radiological Consequences of a Fuel Handling Accident," WCAP-7518-L (Proprietary), July 1971 and WCAP-7828 (Nonproprietary), December 1971.

TABLE 15.7-17

RADIONUCLIDE CONCENTRATION

Isotope	Concentration In Marsh (μCi/ml)	Concentration In Hampton Harbor (μCI/ml)
Sr-90	1.7E-08*	6.9E-14
Y-91	1.1E-03	4.6E-09
Zr-95	2.0E-04	8.2E-10
Nb-95	1.7E-05	6.9E-11
Cs-137	6.9E-04	2.8E-09
Ce-144	1.8E-03	7.4E-09
Mn-54	3.4E-03	1.4E-08
Co-60	6.2E-03	2.5E-08
Fe-59	6.6E-05	2.7E-10
Cr-51	3.0E-06	1.2E-11
H-3	8.0E-01	3.3E-06

* 1.7E-08 = 1.7×10^{-8}

TABLE 15.7-18
(Sheet 1 of 3)

SUMMARY OF PARAMETERS AND ASSUMPTIONS
USED FOR FUEL HANDLING ACCIDENT

Parameters	Design Basis Assumptions	Realistic Assumptions
I. Source Data:		
Reactor Core Power Level, MWt	3654[1]	3654
Rods Array in Assembly	17x17	17x17
Rods in Assembly	264	264
No. of Fuel Assemblies	193	193
Total No. of Fuel Rods in Core	50,952	50,952
No. of Damaged Fuel Rods	264	264
Burnup (MWd/Mtu)	Equilibrium Cycle	Equilibrium Cycle
Decay Time, Hrs	100	100
Radial Peaking Factor	1.65	1.3
Activity Released Data:		
Minimum Water Depth Above Damaged Fuel Rods, Ft	23/10	23
Maximum Fuel Rod Release Pressure, psig	≤1200	≤1200
Peak Linear Power Density for Highest Power Assembly Discharged, kw/ft	13.9	13.9
Maximum Centerline Operating Fuel Temperature for Assembly, °F	3550	3550
Pool Decontamination Factor for for Noble Gases	1	1
Effective Pool Decontamination Factor for Iodine	100/37	760

(1) The gap release fractions are based on operation at 3411 MWth

TABLE 15.7-21

ACTIVITY RELEASED FROM WATER SURFACE AND MAXIMUM BUILDING AIR CONCENTRATION - CONSERVATIVE CASE

Radionuclide	Release From Water Surface (Ci)	Max. Containment Air Concentration[a] (μCi/cc)	Max. Fuel Storage Bldg. Air Concentration[b] (μCi/cc)
I-131	6.3E+02*	8.2E-03	7.4E-02
I-133	6.8E+01	8.8E-04	8.0E-03
I-135	5.0E-02	6.5E-07	5.9E-06
Kr-85	2.3E+03	3.0E-02	2.7E-01
Xe-131m	6.8E+02	8.8E-03	8.0E-02
Xe-133m	7.8E+03	1.0E-01	9.2E-01
Xe-133	1.3E+05	1.7E+00	1.5E+01
Xe-135m	8.0E-01	1.0E-05	9.4E-05
Xe-135	2.4E+02	3.1E-03	2.8E-02

* 6.3E+02 = 6.3×10^2

[a] Containment Air Volume: 2.715×10^6 ft^3

[b] Fuel Storage Building Air Volume: 3.0×10^5 ft^3 - Dropped assembly case

TABLE 15.7-22

ACTIVITY IN HIGHEST RATED ASSEMBLY
AT TIME OF REACTOR SHUTDOWN AND
100 HOURS AFTER SHUTDOWN - REALISTIC CASE

Radionuclide	Total Activity At Shutdown (Ci)	Fraction In Cladding Gap	Activity in Cladding Gap (Ci) At Reactor Shutdown	Activity in Cladding Gap (Ci) 100 Hours After
I-131	7.1E+05*	0.0186	1.3E+04	9.1E+03
I-132	9.8E+05	0.0021	2.1E+03	Negl.**
I-133	1.5E+06	0.0068	1.0E+04	3.6E+02
I-134	1.6E+06	0.0013	2.1E+03	Negl.
I-135	1.4E+06	0.0038	5.4E+03	1.5E-01
Kr-83m	---	---	---	---
Kr-85m	2.0E+05	0.0053	1.0E+03	Negl.
Kr-85	6.1E+03	0.273	1.7E+03	1.7E+03
Kr-87	3.5E+05	0.0015	5.2E+02	Negl.
Kr-88	5.0E+05	0.0053	2.7E+03	Negl.
Xe-131m	5.0E+03	0.0227	1.1E+02	1.1E+02
Xe-133m	2.1E+05	0.0102	2.1E+03	6.1E+02
Xe-133	1.4E+06	0.0153	2.1E+04	1.4E+04
Xe-135m	2.9E+05	0.00070	2.0E+02	2.4E-02
Xe-135	3.0E+05	0.0042	1.3E+03	7.3E+00
Xe-138	1.3E+06	0.00074	9.6E+02	Negl.

* 7.1E+05 = 7.1×10^5
** Negligible (<0.1 Ci)

TABLE 15.7-23

ACTIVITY RELEASED FROM WATER SURFACE AND
MAXIMUM BUILDING AIR CONCENTRATION - REALISTIC CASE

Radionuclide	Release From Water Surface (Ci)	Max. Containment Air Concentration[a] (μCi/cc)	Max. Fuel Storage Bldg. Air Concentration[b] (μCi/cc)
I-131	1.2E+01*	1.6E-04	1.4E-03
I-133	4.7E-01	6.1E-06	5.5E-05
I-135	2.0E-04	2.6E-09	2.3E-08
Kr-85	1.7E+03	2.2E-02	2.0E-01
Xe-131m	1.1E+02	1.5E-03	1.3E-02
Xe-133m	6.1E+02	7.9E-03	7.1E-02
Xe-133	1.4E+04	1.8E-01	1.6E+00
Xe-135m	2.4E-02	3.1E-07	2.8E-06
Xe-135	7.3E+00	9.5E-05	8.6E-04

* 1.2E+01 = 1.2×10^{1}

a Containment Air Volume: 2.715×10^{6} ft^3

b Fuel Storage Building Air Volume: 3.0×10^{6} ft^3 - Dropped assembly case

TABLE 15.7-24

ACTIVITY RELEASED TO ENVIRONMENT
FROM FUEL STORAGE BUILDING*

Radionuclide	Conservative Case (Ci)	Realistic Case (Ci)
I-131	3.9E+01**	1.2E-01
I-133	4.2E+00	4.7E-03
I-135	3.1E-03	2.0E-06
Kr-85	2.3E+03	1.7E+03
Xe-131m	6.9E+02	1.1E+02
Xe-133m	7.8E+03	6.1E+02
Xe-133	1.3E+05	1.4E+04
Xe-135m	4.5E+00	1.4E-01
Xe-135	2.4E+02	7.3E+00

** 3.9E+01 = 3.9×10^{1}
* Case 1 - dropped assembly

TABLE 15.7-25

SUMMARY OF OFFSITE DOSES DUE TO
FUEL HANDLING ACCIDENT IN FUEL STORAGE BUILDING

Site	Conservative Case (rem)			Realistic Case (rem)		
	Thyroid	Whole Body	Skin	Thyroid	Whole Body	Skin
Case 1 - Dropped Assembly						
EAB (0-2 hours)	5.5E+00	3.5E-01	9.0E-01	2.4E-03	5.1E-03	1.5E-02
LPZ (Duration of Accident)	2.3E+00	1.4E-01	3.7E-01	6.5E-04	1.4E-03	4.2E-03
Case 2a - (damaged assembly with 10 feet of water above release point) R. G. 1.109 DCF's						
EAB 0-2 Hr	1.3E+01	3.5E-01	9.0E-01		N/A	
Case 2b - (damaged assembly with 10 feet of water above release point) ICRP-30 DCF's						
EAB 0-2 Hr	9.6E+00	5.1E-01	7.3E-01		N/A	

* 3.5E-01 = 3.5×10^{-1}
EAB: Exclusion Area Boundary; LPZ: Low Population Zone

TABLE 15.7-25a

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES DUE TO FUEL HANDLING ACCIDENT IN CONTAINMENT BUILDING

	DOSE (REM)		
Site	Thyroid	Whole Body	Skin
EAB (0-2 hours)	6.3E+01	2.0E+00	5.0E-01
CONTROL ROOM (0-30 Days)	6.7E+00	3.0E-01	1.5E+00

* 3.5E-01 = 3.5×10^{-1}

EAB: Exclusion Area Boundary; LPZ: Low Population Zone

Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

CHAPTER 17

QUALITY ASSURANCE

CONTENTS

CHAPTER 17

QUALITY ASSURANCE

TABLES

a. Executive Vice President and Chief Nuclear Officer (Seabrook)

The Executive Vice President and Chief Nuclear Officer (Seabrook) has overall responsibility for implementation of the OQAP at Seabrook Station, including plant operation; maintenance, design/engineering, licensing and support services. He has assigned responsibility for these activities to the Station Director, Director of Engineering, and Director of Support Services.

b. Station Director

The Station Director - Seabrook is responsible for the operation and operational support of Seabrook Station. Seabrook Station is headed by the Station Director who is responsible for the operation and administration of Seabrook Station. He has overall responsibility for implementation aspects of the Program and is Chairman of the Station Operation Review Committee (SORC). To carry out this assignment, the Station Director has the staff and organization described in Chapter 13. The Nuclear Training Manager reports directly to the Station Director. The Training Department will remain independent of the remainder of the Seabrook Station staff to ensure that training is able to maintain sufficient organizational freedom to allow independency from operating pressures. The various parts of this organization implement their assigned aspects of the Program as follows:

1. Assistant Station Director

The Assistant Station Director reports directly to the Station Director and is responsible for

(a) being a member of the Station Operation Review Committee (SORC), and

(b) assuming the responsibilities of the Station Director in his/her absence.

2. Chemistry and Health Physics Manager

The Chemistry and Health Physics Manager reports directly to the Station Director and is responsible for

(a) ensuring that the quality of steam and water is within specifications, and

(b) health physics, radiation protection and radwaste control programs at the Station.

3. Maintenance Manager

The Maintenance Manager reports directly to the Station Director and is responsible for

(a) performing the support functions that include the corrective and preventive maintenance programs, maintenance related surveillance activities, and Station modification and repair actions, and

(b) scheduling the performance of work and controlling the material, personnel and processes involved.

4. Operations Manager

The Operations Manager reports directly to the Station Director and is responsible for

(a) operating equipment at the Station in compliance with Technical Specifications and other license requirements,

(b) assisting in the training of operations personnel to assure an adequate number of qualified employees for each task,

(c) preparing, reviewing, approving and implementing the operating procedures to be used for Station operations,

(d) directing actions, within the realm of the Operations Group, to perform the balance of the surveillance testing program required by the Station license, and

(e) maintaining a staff of fire protection personnel.

5. Work Control and Outage Manager

The Work Control and Outage Manager reports directly to the Station Director and is responsible for

(a) long-range planning and scheduling of refueling and planned maintenance outage planning and scheduling and outage coordination.

(b) forced outage planning and scheduling,

(c) daily work scheduling,

(d) work status tracking and Work Control Program trend reporting.

6. Nuclear Training Manager

The Training Manager reports directly to the Station Director and is responsible for direction and control for the conduct of training at Seabrook Station. He has overall responsibility for administrative activities, program development and evaluation, record keeping and meeting accreditation requirements.

c. Director of Support Services

The Director of Support Services, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), is responsible for Station security services, emergency preparedness, community and public affairs, materials management and labor relations, information technology, performance improvement and business services and facilities. The Director of Support Services is also responsible for administering the overall corrective action program, trend analysis and failure prevention techniques.

d. Environmental, Government and Owner Relations Manager

The Environmental, Government and Owner Relations Manager, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), is responsible for the overall direction of nonradiological environmental compliance, hazardous waste, government affairs, and industry relations at Seabrook Station.

e. Director of Engineering

The Director of Engineering, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), is responsible for North Atlantic engineering design, engineering programs, reactor engineering, shutdown risk management, configuration management, plant engineering and regulatory programs, including NRC and generic licensing. To assist in the accomplishment of these responsibilities, various Seabrook Station managers report to the director.

f. Nuclear Oversight Manager

The Nuclear Oversight Manager, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), has overall responsibility for assuring that the Operational Quality Assurance Program is effectively implemented by all organizations performing work on systems and equipment within the scope of the OQAP through the performance of audits, reviews, inspections and surveillances.

Oversight has the freedom and authority to perform independent reviews of OQAP-related work and request work stoppages or remedial actions if conditions adverse to quality are encountered. Functions to be performed by Oversight are matched with available manpower resources on a short- and long-term basis. For short-term inspection planning, specific work packages are developed. Due to the time required to recruit and train personnel and other variables in the work load, certain tasks may be assigned to consultants as needed. Oversight authority extends over the North Atlantic Corporate Staff, the Seabrook Station Staff and any other organization performing OQAP-related work for Seabrook Station. The Nuclear Oversight Manager has a line of communication with the directors and managers of the staffs to assure a timely resolution of OQAP-related problems, findings, and corrective actions. The communication interface is shown in Figure 17.2-1. The qualifications and experience of the Nuclear Oversight Manager are as defined in ANSI/ANS 3.1-1978 for Professional-Technical. The Nuclear Oversight Manager directs and supervises quality activities such that he/she

- approves the Program and all changes to the Program,
- maintains communications with NU and North Atlantic directors and managers, and other appropriate personnel, as required, to maintain cognizance of matters relating to implementation of the Program,

- directs Oversight in the implementation of the OQAP,
- develops a training program for Oversight personnel and provides for their certification, as required,
- identifies quality problems and evaluates their extent and safety implications,
- recommends, provides or initiates solutions to identified quality problems,
- verifies implementation of approved solutions to such problems, and
- administers the audit, surveillance and inspection programs of the North Atlantic organization.

Oversight also provides the onsite verification and assessment of the Program implementation through the performance of selected reviews, inspections and surveillances.

The Nuclear Oversight Manager receives prior notification of SORC meetings. The audit and surveillance process will perform oversight to ensure SORC activities are performed consistent with Technical Specifications. Oversight is on distribution for pertinent meeting notices, correspondence and information, and a representative attends any meeting which appears appropriate. Oversight personnel routinely attend and participate in work schedule and status meetings to assure that they are kept abreast of day-to-day work assignments throughout the plant, and that there is adequate Oversight coverage relative to procedural and inspection controls, acceptance criteria, and Oversight staffing and qualification of personnel to carry out Oversight assignments.

g. Concerns Resolution Manager

The Concerns Resolution Manager, reporting to the Nuclear Oversight Manager is responsible for the administration of the nuclear safety concerns program.

These sections report to the Nuclear Oversight Manager.

The Nuclear Oversight Manager and his/her staff embody the necessary technical and professional qualifications and expertise, and are responsible for

1. assisting the Training Department in providing QA Training for North Atlantic personnel,
2. training, retraining, and qualifying personnel in the specific Oversight skills and techniques required to perform, audits, surveillances and inspections of OQAP-related work associated with North Atlantic operations, maintenance and other activities,
3. monitoring and reviewing the conduct of Oversight activities and the performance of inspections,
4. performing Oversight inspection functions not delegated to other groups/organizations,
5. performing audits and surveillances of Station programs and activities within the scope of the OQAP,
6. reviewing to assure that contractors or service agencies performing onsite work, within the scope of the OQAP, employ adequate QA programs and implementing procedures,
7. exercising stop-work authority, and
8. evaluating and reporting on OQAP effectiveness.

h. Managers of Design Engineering

The Managers of Design Engineering are responsible for North Atlantic engineering design and engineering programs (e.g., equipment qualification, reactor engineering/fuel design, inservice inspection, and fire protection). The managers report to the Director of Engineering and interface with the Nuclear Oversight organization.

i. Manager of Plant Engineering

The Manager of Plant Engineering, reporting to the Director of Engineering, is responsible for plant engineering functions including component diagnostics, plant engineering, and predictive maintenance technologies and interfaces with the Nuclear Oversight organization.

j. Regulatory Programs Manager

The Regulatory Programs Manager reports to the Director of Engineering and interfaces with the Nuclear Oversight Manager. The manager is responsible for the management of NRC licensing compliance and generic licensing activities, reliability and safety engineering, records management and configuration control.

k. Manager of Materials Management and Labor Relations

The Manager of Materials Management and Labor Relations reports to the Director of Support Services and interfaces with the Nuclear Oversight Manager at Seabrook. He/she is responsible for inventory, procurement engineering, purchasing, contracts, and receipt inspection.

l. Manager of Performance Improvement and Project Management

The Manager of Performance Improvement and Project Management reports to the Director of Support Services and is responsible for administering the overall corrective action program, trend analysis and failure prevention techniques. This position interfaces with the Nuclear Oversight Manager for Corrective Action program implementation and effectiveness.

m. Security Manager

The Security Manager reports directly to the Director - Support Services and is responsible for administering and monitoring performance of Station security services.

n. Contractor Support

Provides technical expertise in areas such as plant engineering, licensing, environmental engineering, fuel cycle, and nuclear engineering as well as nuclear oversight services as directed by contract.

o. Nuclear Safety Audit Review Committee (NSARC)

NSARC is an executive body that is responsible for conducting a critical examination of Station activities, including Station operation, evaluation of procedures, investigations of abnormal conditions, and functioning of the OQAP. UFSAR, Chapter 17, Appendix 17C defines the responsibilities and authority of NSARC. A written charter, approved by the Executive Vice President and Chief Nuclear Officer, designates the membership authority and rules for conduct of activities.

p. Station Operation Review Committee (SORC)

SORC is an advisory group, composed of station management and supervisory personnel, for the purpose of reviewing current activities and determining the effect on operational safety. SORC recommends to the Station Director approval or disapproval of proposals considered by the Committee. Technical Specifications define the responsibilities and authority of SORC. A written charter, approved by the Station Director, designates membership, authority and rules for conduct of activities.

h. Spent Fuel Pool Liner

Backfits, repairs or modifications to the spent fuel pool liner will be conducted under the scope of the OQAP.

i. Flood Prevention Design Features

Modifications of the site and roof drainage systems, the seawall, retaining walls, and other revetments surrounding the plant will be evaluated to determine if their implementation will increase the flood vulnerability of safety-related items. Those modifications determined to affect safety-related items will be covered by the OQAP.

All equipment required to cope with Station Blackout is safety-related (see Section 8.4.5). All safety-related equipment is within the scope of the Operational Quality Assurance Program, which complies with the requirements of 10 CFR 50, Appendix B, which exceeds the quality assurance requirements described in RG 1.155.

17.2.2.3 Program Implementation

It is recognized that the degree of Program applicability varies with different systems and activities.

The degree to which the requirements of this Program and its implementing procedures are applied are based upon the following:

- the importance of malfunction or failure of the item to plant safety,
- the potential degradation of a safety-related function as a result of performing an activity,
- the complexity or uniqueness of the item,
- the need for special controls and surveillance or monitoring of processes, equipment and operational activities,
- the degree to which functional compliance can be demonstrated by inspection or test, and
- the quality history and degree of standardization of the item or activity.

A three-level approach is defined to assure program implementation to the degree necessary.

a. Level 1 - This level includes independent inspections and/or tests performed during routine and nonroutine activities by appropriately trained personnel.

b. Level 2 - This level includes surveillance and monitoring activities that are performed by Oversight. Such activities may include observation of tests, inspections and significant activities; review of records and procedures, and verification of test reports.

Records of surveillance and monitoring activities are maintained.

c. Level 3 - This level includes a comprehensive audit and evaluation program initiated by the Audit Manager. An outside organization is utilized to assure proper functioning of Levels 1, 2 and 3. This level includes measures performed to verify that activities required by the OQAP are established, implemented and satisfy regulatory requirements.

17.2.2.4 Program Standards

North Atlantic maintains a working knowledge of applicable industry codes, standards, federal regulations and guides. The OQAP, complies with the following references and the regulatory position of the Regulatory Guides, except as noted in Appendix 17A.

a. 10 CFR, Part 50, Appendix A - General Design Criteria For Nuclear Power Plants.

b. 10 CFR, Part 50, Appendix B - Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants.

c. 10 CFR, Part 50.55a Codes and Standards.

d. 10 CFR, Part 50.59 Changes, Tests and Experiments.

e. 10 CFR, Part 71, Subpart H - Quality Assurance Criteria For Shipping Packages For Radioactive Material.

f. 10 CFR, Part 50.34 (b.6.ii) Final Safety Analysis Report.

g. Regulatory Guide 1.8, (Rev. 2, April 1987, Personnel Selection and Training (endorses ANSI N18.1-1971; however, ANS 3.1-1978 will be the standard used). See Section 1.8 for details on position using ANSI/ANS 3.1-1981.

h. Regulatory Guide 1.26, Rev. 3, February 1976, Quality Group Classification, and Standards For Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants.

nonconformance reporting system. Each report of a nonconforming item is reviewed. This review verifies that the deficiency was properly and correctly stated, the disposition and corrective action are acceptable, and assesses the effectiveness of the steps to prevent recurrence.

17.2.15.3 Trend Analysis

The Director of Support Services maintains a system to recognize, evaluate, document, and assess quality trends. The system provides for periodic analysis of reports on nonconforming items, adverse condition reports, and the submittal of significant results to the Executive Vice President and Chief Nuclear Officer (Seabrook) and appropriate NU and North Atlantic management personnel for review and appropriate action.

17.2.16 Corrective Action

17.2.16.1 Initiation

Corrective action is that action taken to identify, correct and preclude recurrence of conditions adverse to the quality of activities or equipment. North Atlantic programs identify those conditions for which corrective action may be warranted including

a. failure of a structure, system or component that is within the scope of the OQAP,

b. defect of an item or service that could, if uncorrected, lead to failure or malfunction,

c. operation outside of specified limits,

d. repetitive minor problems which may be symptomatic of a larger problem,

e. reportable occurrences as defined by the Technical Specifications,

f. loss or apparent loss of special nuclear material (SNM), and

g. significant conditions identified by the NRC, SORC, NSARC or audit program.

Corrective action is normally documented through appropriate procedures. In the case of significant conditions adverse to safety, the corrective action includes an evaluation of the cause of the condition, the recommended action to prevent or reduce the probability of recurrence, and verification of completion of corrective action. A special report may be prepared when a significant condition adverse to safety is identified. This report identifies root causes and documents action taken to preclude recurrence.

17.2.16.2 Assessment

Procedures require that corrective action associated with nonconformance reports, and other special corrective action reports are reviewed for adequacy and timeliness. The Seabrook Nuclear Oversight Manager periodically reports to the Station Director and Executive Vice President and Chief Nuclear Officer (Seabrook) on the effectiveness of the corrective action process and status of incomplete items. Corrective action program results are included in the trend analysis program.

17.2.17 Quality Assurance Records

17.2.17.1 Identification

Cognizant managers have the responsibility for determining and identifying quality assurance records that are to be retained and their retention period. Examples of the types of records retained include procurement documents, procedures, NDE results; inspection, audit and test results; material analyses; equipment, process and personnel qualifications; calibration records, nonconformances and corrective action results and station operating records. Inspection and test records, where applicable, identify the inspector or data recorder, type of observation, results, acceptability of the results, date, and action taken in connection with any deficiencies noted. The compilation of records generated is forwarded to the Document Control Center for inclusion into the Station Records Management System. The system is compatible with the design and construction phase records system.

17.2.17.2 Receipt, Storage and Retrieval

Station procedures identify the responsibility of personnel and actions required to control the receipt, storage and retrieval of quality assurance records. A suitable storage facility, designed to prevent loss or deterioration of quality assurance records, is permanently located onsite. Records, whether original or copies, are indexed, filed and maintained to aid in the retrieval process.

17.2.17.3 Supplier Records

Principal suppliers, their sub-tier suppliers and other suppliers are required to identify quality assurance records generated throughout the life of the contract in accordance with the appropriate provisions of the North Atlantic procurement documents. The suppliers are required to maintain a record system and, upon completion of the contract, either continue maintaining the records or forward them to North Atlantic for incorporation into the North Atlantic Records Management System. Internal procedures identify the receipt, inspection and transmittal activities and responsibilities associated with supplier records.

17.2.18 Audits

17.2.18.1 Planning

The Nuclear Oversight Manager is responsible for development and management of an audit program pertaining to activities associated with operation and operational support at Seabrook Station. The Plant Support Oversight and Audit Manager, reporting to the Nuclear Oversight Manager, is responsible for scheduling, coordinating and performing audits using Audit personnel who normally report to other Oversight supervisors. Formal reports are distributed by the Plant Support Oversight and Audits Manager to other management positions as required by established procedures. A plan, which identifies the audits to be performed and their frequency, is approved by the Plant Support Oversight and Audits Manager. The audit plan is based on the status performance and safety significance of activities being performed and ensures that an audit of all functions is completed within a two-year period. These are identified and included in the plan which is updated semiannually. Additional audits may be scheduled when conditions warrant, i.e., extensive reorganization, quality becomes suspect, or supplier implementation of the QA program is suspect.

17.2.18.2 Performance

Audits, based on the pre-established schedule, are performed by trained and qualified personnel using appropriate procedures, instructions and checklists. The procedures, instructions and checklists provide a basis for performance of audits including pre-audit and post-audit conferences and the mechanics of the audit process. The mechanics of the process include an objective evaluation of practices, procedures, instructions, activities and items, and review of documents and records to determine the extent that the quality assurance program is effective and is properly implemented. Auditors do not have direct responsibility in the area being audited. Their qualifications, as a minimum, are based on prior pertinent experience, specialized training and education in accordance with applicable procedures. The audits conducted on site are performed under the direction of the Audit Team Leader. In addition the NSARC may request audits, which may be performed by the Oversight Group or contracted to outside professionals.

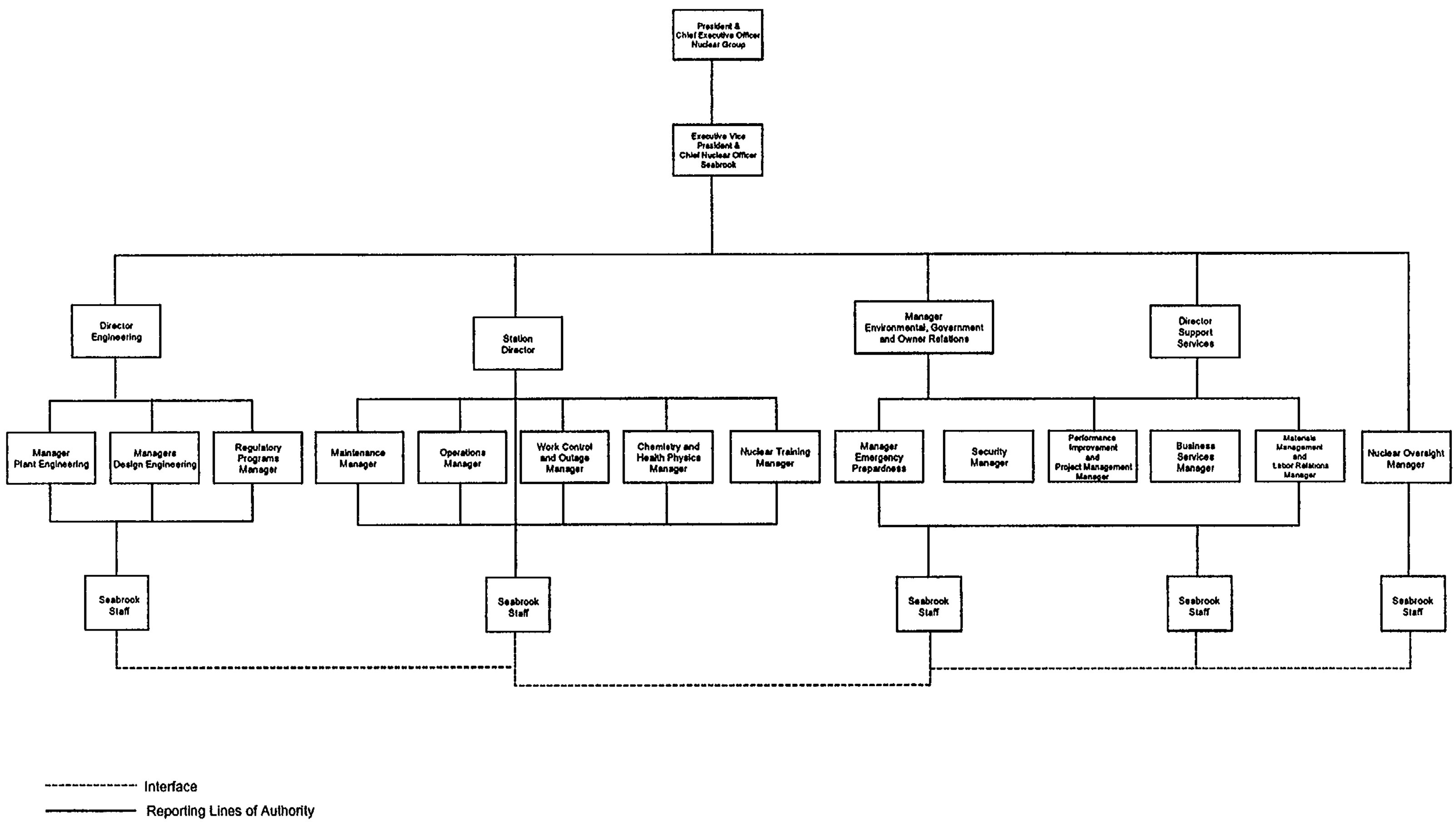
The Oversight Group will perform independent audits under the direction of the NSARC. The audits will be conducted in accordance with Section 17.C of the UFSAR and may involve contracting outside professionals.

17.2.18.3 Reporting and Follow-up

An audit report is generated at the completion of each audit and submitted to the audited manager of the functional area, the NSARC Chairman and to other appropriate management personnel. NSARC also receives a copy or summary report for their review and assessment of the audit program. Follow-up is required by both the audited and auditing organizations when deficiencies are identified. The audited organization is responsible to review and investigate the nature and cause of the deficiency and to provide appropriate corrective action. The Audit Team Leader is responsible for evaluation of proposed or completed corrective action and confirmation of satisfactory accomplishment.

17.2.18.4 Audit Program Review

An independent audit of the QA program effectiveness and appropriateness is initiated biennially by NSARC in accordance with Section 17.C of the UFSAR.



G:\Word\Images_P\Images_P\UFSAR.1721.ds4

Seabrook Station Updated Final Safety Analysis Report	Nuclear Quality Assurance Interface	
	Rev. 07	Figure 17.2-1

"approved instruction manuals" shall be interpreted to mean the manuals supplied by the vendor as required by the procurement order. These manuals may be reviewed by NAESCO in part or in total to assure proper operational sequence and adequacy of acceptance criteria. (3) No special checks will be made by the person withdrawing a replacement part from the warehouse-equivalent controls are assured by compliance set forth in the Operational Quality Assurance Program; and, (4) will be complied with, as stated, by individual technicians as part of the maintenance/modification process.

b. With regard to Section 4 of ANSI N45.2.4-1972 titled Installation: Will be implemented by inclusion, as necessary, in the appropriate maintenance or modification procedure, where such procedures are used. Standard NAESCO maintenance practices require that care be exercised in the six areas listed whether a procedure is required or not.

c. With regard to Section 6.2.1 of ANSI N45.2.4-1972 titled Equipment Tests: The last paragraph of this section deals with tagging and labeling. NAESCO will comply with an alternate last paragraph which reads: "Each safety-related item of process instrumentation is identified with a unique number. This number is utilized in instrument maintenance records so that current calibration status, including data such as the date of the calibration and the person performing the calibration, can be readily determined. Such information may also be contained on tags or labels which may be attached to installed instrumentation."

Regulatory Guide 1.33 (Rev. 2, 2/78) — Quality Assurance Program Requirements (Operations)

Endorses ANSI N18.7-1976/ANS 3.2

During the operational phase, the Operational Quality Assurance Program includes and complies with this guide with the following clarification:

a. Paragraph C.4 of Regulatory Guide 1.33 (and Section 4.5 of ANSI N18.7 which it references) states in part that audits shall be performed with the frequency commensurate with their safety significance and in such a manner to ensure that an audit of all safety-related functions is completed within a period of 2 years. The paragraph continues with a recommendation to include the following audit elements: (1) the results of actions taken to correct deficiencies every 6 months; (2) conformance to the technical specifications and license conditions every 12 months, and (3) performance, training, and qualification of the facility staff every 12 months. These audit elements are included in many audits conducted at Seabrook Station. Exception is taken to the specified periodicity. These audit elements, as with all elements, will be scheduled based on safety significance and performance, as stated in the first sentence, subject to the maximum allowable extension stated in UFSAR Section 17.C.8.

b. Paragraph C.5.a of Regulatory Guide 1.33 (and Section 4.4 of ANSI N18.7 which it references) will be implemented with the clarification that "periodically" (as used in the second paragraph, first sentence) shall not require a set frequency, and "documented" (as used in the second paragraph, second sentence) shall not require typed minutes, but may mean hand written notes or memoranda.

c. Paragraph C.5.d of Regulatory Guide 1.33 (and Section 5.2.7.1 of ANSI N18.7 which it references) will be implemented by adding the clarifying phrase "Where practicable" in front of the fourth sentence of the fifth paragraph. The Regulatory Guide's changing of the two uses of the word "should" in this sentence to "shall" unnecessarily restricts NAESCO's options on repair or replacement parts. It is not always practical to test parts prior to use.

For modifications where these requirements are not considered practical, a review in accordance with the provisions of 10 CFR 50.59 will be conducted and documented.

d. Paragraph C.5.f of Regulatory Guide 1.33 (and Section 5.2.19 (2) of ANSI N18.7 which it references) will be implemented with the substitution of the word "feasible" for the word "possible" in the last sentence.

e. With regard to Section 5.2.7 of ANSI N18.7-1976 titled Maintenance and Modification: Since some emergency situations could arise which might preclude preplanning of all activities, NAESCO will comply with an alternate to the first sentence in the second paragraph which reads: "Except in emergency or abnormal operating conditions where immediate actions are required to protect the health and safety of the public, to protect equipment or personnel, or to prevent the deterioration of plant conditions to possible unsafe or unstable level, maintenance and modification of equipment shall be preplanned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances which conform to applicable codes, standards, specifications and criteria. Where written procedures, documented instructions, or drawings would be required and are not used, the activities that were accomplished shall be documented after the fact and receive the same degree of review as if the work were preplanned. Skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineations in a written procedure."

f. With regard to Section 5.2.7.1 of ANSI N18.7-1976 titled Maintenance Programs: NAESCO will comply with the requirements of this section with the clarification obtained by adding the phrase "Where feasible" before the first sentence of the fifth paragraph. It is not always possible to promptly determine the cause of a malfunction. In all cases, NAESCO will initiate proceedings to determine the cause, and will make such determinations promptly, where it is feasible.

g. With regard to Section 5.2.10 of ANSI N18.7-1976 titled Housekeeping and Cleanliness: Instead of the pre-closure cleanliness inspection specified in the second paragraph, a documented cleanliness verification will be performed prior to closure of any portion of safety-related systems which may be subject to potential contamination with foreign material. The personnel performing this verification will be qualified to the requirements of ANSI/ANS 3.1-1978. Additionally, an independent pre-closure cleanliness inspection will be performed at a frequency commensurate with the safety significance of the affected system or component.

h. With regard to Section 5.2.15 of ANSI N18.7-1976 titled Review, Approval, and Control of Procedures: NAESCO will comply with the requirements of this section with the exception of the requirement to review plant procedures no less frequently than every two years to determine if changes are necessary or desirable. Other programs in place at Seabrook Station make the biennial review redundant. The effectiveness of these programs in maintaining procedures current is verified by a system of periodic audits, surveillances, and/or inspections.

NAESCO implements programmatic controls which specify conditions where mandatory review of plant procedures will apply. These conditions include reviewing applicable procedures following an unusual incident such as an accident, an unexpected transient, significant operator error, or equipment malfunction and following any modification to a system.

NAESCO performs biennial reviews of non-routine Emergency Operating Procedures, Abnormal Operating Procedures and the Emergency Plan Procedures. This biennial review requirement is included in the administrative controls for these procedures.

i. With regard to Section 5.2.16 of ANSI N18.7-1976 titled Measuring and Test Equipment: NAESCO will comply with the requirements of this section with the clarification that the first paragraph delineates the requirements for installed instrument and control devices. These installed Instrument and Control devices will be calibrated by the use of a transfer standard which meets the requirements of this section.

a. With regard to Section 1.2.2 of ANSI N45.2.13-1976 titled Purchaser's Responsibilities: (a) Item d is interpreted to mean that NAESCO will assure that the appropriate requirements of the Operational Quality Assurance Program will be implemented. ANSI N45.2 (referenced by this item) is not included in the NAESCO Operational Quality Assurance Program, although equivalent controls are established by way of the program's commitments to other ANSI standards and regulatory Guides.

b. With regard to Section 3.2.2 of ANSI N45.2.13-1976 titled Technical Requirements: The first sentence is revised to read: "Technical Requirements shall be specified in the procurement documents and/or, where appropriate, by reference to or inclusion of the specific drawings, specifications, codes, regulations, procedures or instructions including revisions thereto that describe the items or services to be furnished."

c. Exception is taken to the commitment of regulatory Guide 1.123 Revision 1, position C.6.e that revises Section 10.3.4 of ANSI N45.2.13 to require that "post installation test requirements and acceptance documentation shall be mutually established by the purchaser and supplier." NAESCO will comply with the requirements of Section 10.3.4 of ANSI N45.2.13 as modified by Generic Letter 89-02. The modification endorses the guidelines contained in EPRI NP-5652 for commercial grade items, specifically stating component manufacturers may be contacted to obtain information for verification of components.

Regulatory Guide 1.144 (Rev. 1, 9/80) — Auditing or Quality Assurance Programs for for Nuclear Power Plants

Endorses ANSI N45.2.12-1977

During the Operations phase, the Operational Quality Assurance Program includes and complies with this guide with the following clarification:

a. With regard to Section 3.3 of ANSI N45.2.12-1977 titled Essential Elements of the Audit System: NAESCO will comply with Subsection 3.3.5 revised to read: "Provisions for reporting on the effectiveness of the quality assurance program to the responsible management." Other than audit reports, NAESCO may not directly report the effectiveness of the quality assurance programs to the audited organization when such organizations are outside of NAESCO.

b. With regard to Section 3.5 of ANSI N45.2.12-1977 titled Scheduling: Subsection 3.5.3.1 is interpreted to mean that NAESCO may procedurally control qualification of contractor's or vendor's quality assurance program, prior to awarding a contract or purchase order, by means other than audit.

c. With regard to Section 4.3.1 of ANSI N45.2.12-1977 titled Pre-Audit Conference: NAESCO will comply with the requirements of this section by inserting the word "Normally" at the beginning of the first sentence. This clarification is required because, in the case of certain unannounced audits or audits of a particular operational or work activity, a pre-audit conference might interfere with the spontaneity of the operation or activity being audited. In other cases, persons who should be present at a pre-audit conference may not always be available; such lack of availability should not be an impediment to beginning an audit. Even in the above examples, which are not intended to be all inclusive, the material set forth in Section 4.3.1 will be covered (if considered necessary or desirable) during the course of the audit.

d. With regard to Section 4.4 of ANSI N45.2.12-1977 titled Reporting: NAESCO will comply with Subsection 4.4.3 clarified to read: "Supervisory level personnel with whom major interactions or significant discussions were held during the course of pre-audit (where conducted), audit, and post-audit (where conducted) activities."

Regulatory Guide 1.146 (Rev. 9, 8/80)	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

Endorses ANSI N45.2.23-1978

During the Operations phase, the Operational Quality Assurance Program includes and complies with this guide with the following clarification:

a. With regard to Section 3.2 of ANSI N45.2.23-1978 titled Maintenance of Proficiency: NAESCO will comply with the requirements of this section by defining "annual assessment" as one which takes place every 12±3 months and which used the initial date of certification (not the calendar year) as the starting date for determining when such annual assessments are due.

b. With regard to Section 4.1 of ANSI N45.2.23-1978 titled Organizational Responsibility: NAESCO will comply with this section with the substitution of the following sentence in place of the last sentence in the section: "The Nuclear Oversight Manager or Lead Auditor shall, prior to commencing the audit, assign personnel who collectively have experience or training commensurate with the scope, complexity, or special nature of the activities to be audited."

c. With regard to Section 5.3 of ANSI N45.2.23-1978 titled Updating of Lead Auditor's Records: NAESCO will substitute the following sentence for this section: "Records for each Lead Auditor shall be maintained and updated during the period of the annual management

APPENDIX 17C

NUCLEAR SAFETY AUDIT REVIEW COMMITTEE

17C NUCLEAR SAFETY AUDIT REVIEW COMMITTEE (NSARC)

17C.1 FUNCTION

The NSARC shall function to provide independent review and audit of designated activities. The NSARC shall report to and advise the Executive Vice President & Chief Nuclear Officer on those areas of responsibility specified in Specifications 6.4.3.7 and 6.4.3.8.

17C.2 COMPOSITION

The NSARC shall be composed of at least five (5) individuals. The Chairman, Vice Chairman and members, including designated alternates, shall be appointed in writing by the Executive Vice President & Chief Nuclear Officer. Collectively, the individuals appointed to the NSARC should have experience and expertise in the following areas:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

Each member shall meet the qualifications of ANSI 3.1-1978, Section 4.7.

17C.3 ALTERNATES

All alternate members shall be appointed in writing by the Executive Vice President & Chief Nuclear Officer to serve on a temporary basis; however, no more than a minority shall participate as voting members in NSARC activities at any one time.

17C.4 CONSULTANTS

Consultants shall be utilized as determined by the NSARC to provide expert advice to the NSARC.

17C.5 MEETING FREQUENCY

The NSARC shall meet at least once per 6 months ± 6 weeks.

17C.6 QUORUM

The quorum of the NSARC shall consist of the Chairman or Vice-Chairman and sufficient NSARC members including alternates to equal at least 50% of the NSARC composition. No more than a minority of the quorum shall have line responsibility for operation of the unit. The Vice Chairman, or his designated alternate, can participate as an NSARC member when the Chairman is in attendance.

17C.7 REVIEW

The NSARC shall be responsible for the review of the following:

a. The safety evaluations for: (1) changes to procedure, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or the Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of station equipment that affect nuclear safety;

g. All Reportable Events;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the SORC.

17C.8 AUDITS

Audits of station activities shall be performed under the cognizance of the NSARC. The audits shall be performed within a 2-year time interval with a maximum allowable extension not to exceed 25% of the specified interval. These audits shall encompass:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the station staff;

c. The results of actions taken to correct deficiencies occurring in station equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The fire protection programmatic controls including the implementing procedures by qualified licensee QA personnel;

f. The fire protection equipment and program implementation utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof;

h. The Offsite Dose Calculation Manual and implementing procedures;

i The Process Control Program and implementing procedures for processing and packaging of radioactive wastes;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring;

k. Any other area of station operation considered appropriate by the NSARC or the Executive Vice President & Chief Nuclear Officer.

17C.9 RECORDS

Records of NSARC activities shall be prepared and distributed as indicated below:

a. Minutes of each NSARC meeting shall be prepared and forwarded to the Executive Vice President & Chief Nuclear Officer within 30 working days following each meeting;

b. Reports of reviews encompassed by Specification 6.4.3.7 shall be included in the minutes where applicable or forwarded under separate cover to the Executive Vice President & Chief Nuclear Officer within 30 working days following completion of the review; and

c. Audit reports encompassed by Specification 6.4.2.8 shall be forwarded to the Executive Vice President & Chief Nuclear Officer and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

LIST OF EFFECTIVE PAGES

This tabulation provides a list of all currently effective pages, tables and figures for volumes 1-13 of the Seabrook Updated FSAR, dated *July 27, 2001*. The Updated FSAR (UFSAR) was Revision 0 according to the requirements of 10 CFR 50.71(e)(3) and all subsequent submittals are updated from this document. A dash in the column after the page, table or figure denotes original issue of this Updated FSAR.

Key
F = figures Att = attachment
T = tables Exh = exhibit
* = The information contained in this appendix was not revised, but has been extracted from the original FSAR and is provided for historical information.

Page No.	Rev. No.
VOLUMES 1 THROUGH 13	
Cover Page	7
Table of Contents	
i	-
ii	-
iii	6
iv	6
v	6
vi	6
vii	6
viii	6
VOLUME 1	
Chapter 1	
1.0-1	4
1.0-2	-
1.0-3	6
1.0-4	-
1.0-5	-
1.0-6	6
1.0-7	-
1.0-8	-
1.0-9	-
1.1-1	4
1.1-2	1
1.1-3	1
T1.1-1 (Sh 1)	7
T1.1-1 (Sh 2)	7
T1.1-1 (Sh 3)	7
T1.1-1 (Sh 4)	7
T1.1-1 (Sh 5)	7
T1.1-1 (Sh 6)	7
T1.1-1 (Sh 7)	7
T1.1-1 (Sh 8)	7
T1.1-1 (Sh 9)	7
T1.1-1 (Sh 10)	7
T1.1-1 (Sh 11)	7
T1.1-1 (Sh 12)	7
T1.1-1 (Sh 13)	7
T1.1-1 (Sh 14)	7
T1.1-1 (Sh 15)	7
T1.1-1 (Sh 16)	7
T1.1-1 (Sh 17)	7
F1.1-1 (Sh 1)	6
F1.1-1 (Sh 2)	4
F1.1-2	4
1.2-1	2
1.2-1a	7
1.2-2	-
1.2-3	6
1.2-4	-
1.2-5	4
1.2-5a	4
1.2-6	2
1.2-7	5
1.2-7a	5
1.2-8	-
1.2-9	-
1.2-10	-
1.2-11	-
1.2-12	-
1.2-13	-
1.2-14	-
1.2-15	5

Rev. Page No.	No.
1.2-16	-
1.2-17	-
1.2-18	-
1.2-19	7
1.2-20	4
F1.2-1 (Sh 1)	7
F1.2-1 (Sh 2)	7
F1.2-2	1
F1.2-3	5
F1.2-4	3
F1.2-5	4
F1.2-6a	6
F1.2-6	4
F1.2-7	-
F1.2-8	4
F1.2-9	-
F1.2-10	-
F1.2-11	-
F1.2-12	-
F1.2-13	-
F1.2-14	-
F1.2-15	5
F1.2-16	5
F1.2-17	-
F1.2-18	-
F1.2-19	-
F1.2-20	-
F1.2-21	1
F1.2-22	-
F1.2-23	4
F1.2-24	-
F1.2-25	-
F1.2-26	1
F1.2-27	-
F1.2-28	-
F1.2-29	-
F1.2-30	-
F1.2-31	5
F1.2-32	4
F1.2-33	-
F1.2-34	-
F1.2-35	5
F1.2-36	-
F1.2-37	5
F1.2-38	5
F1.2-39	4
F1.2-40	-
F1.2-41	-
F1.2-42	-
F1.2-43	5
F1.2-44	-
F1.2-45	-
F1.2-46	4
F1.2-47	-
F1.2-48	-
F1.2-49	3
F1.2-50	-
F1.2-51	-
F1.2-52	-
F1.2-53	-
F1.2-54	-
F1.2-55	-
F1.2-56	4
F1.2-57	5
F1.2-58	5
F1.2-59	4
1.3-1	-
T1.3-1 (Sh 1)	-
T1.3-1 (Sh 2)	-
T1.3-1 (Sh 3)	-

Rev. Page No.	No.
T1.3-1 (Sh 4)	-
T1.3-1 (Sh 5)	-
T1.3-1 (Sh 6)	-
T1.3-1 (Sh 7)	-
T1.3-1 (Sh 8)	-
T1.3-1 (Sh 9)	-
T1.3-1 (Sh 10)	4
T1.3-2 (Sh 1)	4
T1.3-2 (Sh 2)	4
T1.3-2 (Sh 3)	4
T1.3-2 (Sh 4)	4
T1.3-2 (Sh 5)	4
T1.3-2 (Sh 6)	4
T1.3-2 (Sh 7)	4
T1.3-2 (Sh 8)	4
T1.3-2 (Sh 9)	6
T1.3-2 (Sh 10)	4
T1.3-2 (Sh 11)	4
T1.3-2 (Sh 12)	4
T1.3-2 (Sh 13)	4
T1.3-2 (Sh 14)	4
T1.3-2 (Sh 15)	7
T1.3-2 (Sh 16)	4
T1.3-2 (Sh 17)	4
1.4-1	6
1.4-2	-
1.4-3	-
1.5-1	-
1.5-2	-
1.5-3	-
T1.5-1	-
T1.5-2	-
1.6-1	4
T1.6-1 (Sh 1)	4
T1.6-1 (Sh 2)	4
T1.6-1 (Sh 3)	4
T1.6-1 (Sh 4)	4
T1.6-1 (Sh 5)	4
T1.6-1 (Sh 6)	4
T1.6-1 (Sh 7)	4
T1.6-1 (Sh 8)	4
T1.6-1 (Sh 9)	4
T1.6-1 (Sh 10)	4
1.7-1	4
1.7-2	4
1.8-1	-
1.8-2	-
1.8-3	6
1.8-3a	5
1.8-4	6
1.8-5	-
1.8-6	-
1.8-7	6
1.8-8	-
1.8-9	-
1.8-10	-
1.8-11	7
1.8-11a	7
1.8-12	-
1.8-13	-
1.8-14	-
1.8-15	3
1.8-16	4
1.8-17	-
1.8-18	3
1.8-19	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
1.8-20	-
1.8-21	-
1.8-22	6
1.8-23	-
1.8-24	-
1.8-25	-
1.8-26	4
1.8-26a	4
1.8-27	-
1.8-28	3
1.8-28a	4
1.8-29	-
1.8-30	6
1.8-31	5
1.8-31a	5
1.8-32	-
1.8-33	7
1.8-33a	7
1.8-34	-
1.8-35	7
1.8-35a	7
1.8-35b	7
1.8-35c	7
1.8-35d	7
1.8-36	4
1.8-37	7
1.8-37a	6
1.8-37b	7
1.8-37c	7
1.8-38	-
1.8-39	-
1.8-40	-
1.8-41	7
1.8-41a	7
1.8-42	-
1.8-43	-
1.8-44	-
1.8-45	-
1.8-46	-
1.8-47	7
1.8-47a	7
1.8-48	-
1.8-49	6
1.8-49a	6
1.8-50	3
1.8-51	-
1.8-52	-
1.8-53	6
1.8-53a	6
1.8-54	-
1.8-55	5
1.8-55a	5
1.8-56	7
1.8-57	7
1.8-58	7
T1.8-1 (Sh 1)	-
T1.8-1 (Sh 2)	-
T1.8-1 (Sh 3)	-
T1.8-2 (Sh 1)	6
T1.8-2 (Sh 2)	6
T1.8-2 (Sh 3)	6
F1.8-1	-
1.9-1	-
1.9-2	6
1.9-2a	6
1.9-3	3
1.9-4	-
1.9-5	-
1.9-6	-
1.9-7	-
1.9-8	-
1.9-9	-
1.9-10	-
1.9-11	7
1.9-11a	7
1.9-11b	7
1.9-12	-
1.9-13	-
1.9-14	-
1.9-15	-
1.9-16	2
1.9-17	-
1.9-18	-
1.9-19	-
1.9-20	-
1.9-21	-
1.9-22	-
1.9-23	-
1.9-24	-
1.9-25	-
1.9-26	-
1.9-27	6
1.9-28	4
1.9-29	-
1.9-30	-
1.9-31	4
1.9-32	4
1.9-33	-
1.9-34	-

VOLUME 2

Chapter 2

Page No.	Rev. No.
2.0-1	-
2.0-2	2
2.0-3	6
2.0-4	-
2.0-5	-
2.0-6	-
2.0-7	-
2.0-8	-
2.0-9	-
2.0-10	1
2.0-11	-
2.0-12	1
2.0-13	-
2.0-14	-
2.0-15	-
2.0-16	-
2.0-17	-
2.0-18	-
2.0-19	-
2.0-20	1
2.0-21	4
2.0-22	1
2.0-23	1
2.0-24	1
2.0-25	1
2.1-1	5
2.1-2	2
2.1-3	4
2.1-4	4
2.1-5	-
2.1-6	-
2.1-7	-
2.1-8	-
2.1-9	-
2.1-10	-
2.1-11	-
2.1-12	-
2.1-13	-
2.1-14	-
2.1-15	-
2.1-16	-
2.1-17	-
2.1-18	-
2.1-19	-
2.1-20	-
2.1-21	-
2.1-22	-
2.1-23	-
2.1-24	-
2.1-25	-
2.1-26	-
T2.1-1	-
T2.1-2 (Sh 1)	-
T2.1-2 (Sh 2)	-
T2.1-2 (Sh 3)	-
T2.1-2 (Sh 4)	-
T2.1-3	-
T2.1-4 (Sh 1)	-
T2.1-4 (Sh 2)	-
T2.1-4 (Sh 3)	-
T2.1-4 (Sh 4)	-
T2.1-5 (Sh 1)	-
T2.1-5 (Sh 2)	-
T2.1-5 (Sh 3)	-
T2.1-5 (Sh 4)	-
T2.1-5 (Sh 5)	-
T2.1-5 (Sh 6)	-
T2.1-5 (Sh 7)	-
T2.1-5 (Sh 8)	-
T2.1-5 (Sh 9)	-
T2.1-5 (Sh 10)	-
T2.1-5 (Sh 11)	-
T2.1-5 (Sh 12)	-
T2.1-5 (Sh 13)	-
T2.1-6 (Sh 1)	-
T2.1-6 (Sh 2)	-
T2.1-7	-
T2.1-8	-
T2.1-9 (Sh 1)	-
T2.1-9 (Sh 2)	-
T2.1-10	-
T2.1-11	-
T2.1-12 (Sh 1)	-
T2.1-12 (Sh 2)	-
T2.1-13 (Sh 1)	-
T2.1-13 (Sh 2)	-
T2.1-13 (Sh 3)	-
T2.1-13 (Sh 4)	-
T2.1-13 (Sh 5)	-
T2.1-13 (Sh 6)	-
T2.1-13 (Sh 7)	-
T2.1-13 (Sh 8)	-
T2.1-13 (Sh 9)	-
T2.1-13 (Sh 10)	-
T2.1-13 (Sh 11)	-
T2.1.13 (Sh 12)	-
T2.1.13 (Sh 13)	-
T2.1.13 (Sh 14)	-
T2.1-14 (Sh 1)	-
T2.1-14 (Sh 2)	-
T2.1-14 (Sh 3)	-
T2.1-14 (Sh 4)	-
T2.1-14 (Sh 5)	-
T2.1-14 (Sh 6)	-
T2.1-15 (Sh 1)	-
T2.1-15 (Sh 2)	-
T2.1-16	-
T2.1-17	-
T2.1-18	-
T2.1-19 (Sh 1)	-
T2.1-19 (Sh 2)	-
T2.1-19 (Sh 3)	-
T2.1-19 (Sh 4)	-
T2.1-20	-
T2.1-21	-
F2.1-1	-
F2.1-2	-
F2.1-3	-
F2.1-4	-
F2.1-5	6
F2.1-6	-
F2.1-7	-
F2.1-8	-
F2.1-9	-
F2.1-10	-
F2.1-11	-
F2.1-12	-
F2.1-13	-
F2.1-14	-
F2.1-15	-
F2.1-16	-
F2.1-17	-
F2.1-18	-
F2.1-19	-
F2.1-20	-
F2.1-21	-
F2.1-22	-
F2.1-23	-
F2.1-24	-
F2.1-25	-
F2.1-26	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
F2.1-27	-
F2.1-28	-
F2.1-29	-
F2.1-30	-
F2.1-31	-
F2.1-32	-
2.2-1	-
2.2-2	-
2.2-3	-
2.2-4	-
2.2-5	-
2.2-6	-
2.2-7	-
2.2-8	-
2.2-9	-
2.2-10	-
2.2-11	-
2.2-12	-
2.2-13	-
2.2-14	-
2.2-15	-
2.2-16	-
2.2-17	-
2.2-18	-
2.2-19	3
2.2-20	2
2.2-21	2
2.2-22	2
2.2-23	2
2.2-24	2
2.2-25	2
2.2-26	2
2.2-27	2
2.2-28	2
2.2-29	2
2.2-30	2
2.2-31	2
2.2-32	2
2.2-33	2
T2.2-1 (Sh 1)	-
T2.2-1 (Sh 2)	-
T2.2-2 (Sh 1)	-
T2.2-2 (Sh 2)	-
T2.2-2 (Sh 3)	-
T2.2-2 (Sh 4)	-
T2.2-2 (Sh 5)	-
T2.2-3	-
T2.2-4	-
T2.2-5	-
T2.2-6	-
T2.2-7 (Sh 1)	-
T2.2-7 (Sh 2)	-
T2.2-8 (Sh 1)	-
T2.2-8 (Sh 2)	-
T2.2-9	-
T2.2-10	-
T2.2-11	-
T2.2-12	-
T2.2-13	-
T2.2-14	-
T2.2-15	-
T2.2-16	-
T2.2-17	-
F2.2-1	-
F2.2-2	-
F2.2-3	-
F2.2-4	-
F2.2-5	-
F2.2-6	-
F2.2-7	-
F2.2-8	-
F2.2-9	-
F2.2-10	-
F2.2-11	-
F2.2-12	-
F2.2-13	-
F2.2-14	-
F2.2-15	-
2.3-1	-
2.3-2	-
2.3-3	-
2.3-4	3
2.3-5	3
2.3-6	2
2.3-6a	2
2.3-7	1
2.3-8	1
2.3-9	1
2.3-10	3
2.3-10a	3
2.3-11	1
2.3-12	1
2.3-13	1
2.3-14	1
2.3-15	1
2.3-16	1
2.3-17	6
2.3-18	6
2.3-19	1
2.3-20	1
2.3-21	1
2.3-22	1
2.3-23	1
2.3-24	1
2.3-25	1
2.3-26	1
2.3-27	1
2.3-28	1
2.3-29	1
2.3-30	1
2.3-31	1
2.3-32	1
2.3-33	1
2.3-34	1
2.3-35	1
2.3-36	1
2.3-37	1
2.3-38	1
2.3-39	1
2.3-40	1
2.3-41	1
2.3-42	1
2.3-43	1
2.3-44	1
2.3-45	1
2.3-46	1
2.3-47	1
2.3-48	3
T2.3-1 (Sh 1)	-
T2.3-1 (Sh 2)	-
T2.3-1 (Sh 3)	-
T2.3-1 (Sh 4)	-
T2.3-1 (Sh 5)	-
T2.3-2	-
T2.3-3	-
T2.3-4	-
T2.3-5	1
T2.3-6	1
T2.3-7	-
T2.3-8	-
T2.3-9	-
T2.3-10	-
T2.3-11	-
T2.3-12	-
T2.3-13	-
T2.3-14	-
T2.3-15	-
T2.3-16	-
T2.3-17	-
T2.3-18	-
T2.3-19	-
T2.3-20	-
T2.3-21	-
T2.3-22	-
T2.3-23	-
T2.3-24	-
T2.3-25	-
T2.3-26	-
T2.3-27	-
T2.3-28	-
T2.3-29	-
T2.3-30	-
T2.3-31	-
T2.3-32	-
T2.3-33	-
T2.3-34	-
T2.3-35	-
T2.3-36	-
T2.3-37	3
T2.3-38 (Sh 1)	-
T2.3-38 (Sh 2)	-
T2.3-38 (Sh 3)	-
T2.3-39 (Sh 1)	-
T2.3-39 (Sh 2)	-
T2.3-39 (Sh 3)	-
T2.3-40 (Sh 1)	-
T2.3-40 (Sh 2)	-
T2.3-40 (Sh 3)	-
T2.3-40 (Sh 4)	-
T2.3-40 (Sh 5)	-
T2.3-40 (Sh 6)	-
T2.3-41 (Sh 1)	-
T2.3-41 (Sh 2)	-
T2.3-41 (Sh 3)	-
T2.3-41 (Sh 4)	-
T2.3-41 (Sh 5)	-
T2.3-41 (Sh 6)	-
T2.3-42 (Sh 1)	-
T2.3-42 (Sh 2)	-
T2.3-42 (Sh 3)	-
T2.3-42 (Sh 4)	-
T2.3-42 (Sh 5)	-
T2.3-42 (Sh 6)	-
T2.3-43 (Sh 1)	-
T2.3-43 (Sh 2)	-
T2.3-43 (Sh 3)	-
T2.3-43 (Sh 4)	-
T2.3-43 (Sh 5)	-
T2.3-43 (Sh 6)	-
T2.3-44 (Sh 1)	-
T2.3-44 (Sh 2)	-
T2.3-44 (Sh 3)	-
T2.3-44 (Sh 4)	-
T2.3-44 (Sh 5)	-
T2.3-44 (Sh 6)	-
T2.3-45 (Sh 1)	-
T2.3-45 (Sh 2)	-
T2.3-45 (Sh 3)	-
T2.3-45 (Sh 4)	-
T2.3-45 (Sh 5)	-
T2.3-45 (Sh 6)	-
T2.3-46 (Sh 1)	-
T2.3-46 (Sh 2)	-
T2.3-46 (Sh 3)	-
T2.3-46 (Sh 4)	-
T2.3-46 (Sh 5)	-
T2.3-46 (Sh 6)	-
T2.3-47 (Sh 1)	-
T2.3-47 (Sh 2)	-
T2.3-47 (Sh 3)	-
T2.3-47 (Sh 4)	-
T2.3-47 (Sh 5)	-
T2.3-47 (Sh 6)	-
T2.3-48	-
T2.3-49	-
T2.3-50 (Sh 1)	-
T2.3-50 (Sh 2)	-
T2.3-51 (Sh 1)	-
T2.3-51 (Sh 2)	-
T2.3-52 (Sh 1)	-
T2.3-52 (Sh 2)	-
T2.3-53 (Sh 1)	-
T2.3-53 (Sh 2)	-
T2.3-54 (Sh 1)	-
T2.3-54 (Sh 2)	-
T2.3-55 (Sh 1)	-
T2.3-55 (Sh 2)	-
T2.3-56 (Sh 1)	-
T2.3-56 (Sh 2)	-
T2.3-57 (Sh 1)	-
T2.3-57 (Sh 2)	-
T2.3-58	-
T2.3-59	-
T2.3-60	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
T2.3-61	-
T2.3-62	-
F2.3-1	-
F2.3-2	-
F2.3-3	-
F2.3-4	-
F2.3-5	-
F2.3-6	-
F2.3-7	-
F2.3-8 (Sh 1)	-
F2.3-8 (Sh 2)	-
F2.3-9	-
F2.3-10	-
F2.3-11	-
2.4-1	-
2.4-2	-
2.4-3	-
2.4-4	-
2.4-5	-
2.4-6	-
2.4-7	6
2.4-8	6
2.4-9	6
2.4-9a	6
2.4-10	-
2.4-11	-
2.4-12	-
2.4-13	-
2.4-14	-
2.4-15	-
2.4-16	-
2.4-17	-
2.4-18	-
2.4-19	-
2.4-20	-
2.4-21	-
2.4-22	-
2.4-23	-
2.4-24	-
2.4-25	-
2.4-26	-
2.4-27	-
2.4-28	-
2.4-29	-
2.4-30	-
2.4-31	-
2.4-32	6
2.4-32a	6
2.4-33	6
2.4-34	4
2.4-35	-
2.4-36	-
2.4-37	-
2.4-38	-
2.4-39	-
2.4-40	-
2.4-41	-
2.4-42	-
2.4-43	-
2.4-44	-
2.4-45	-
2.4-46	-
2.4-47	-
2.4-48	-
2.4-49	-
2.4-50	-
2.4-51	-
2.4-52	-
2.4-53	-
2.4-54	-
2.4-55	-
T2.4-1	-
T2.4-2	-
T2.4-3	-
T2.4-4	-
T2.4-5	-
T2.4-6	-
T2.4-7	-
T2.4-8	-
T2.4-9	-
T2.4-10	-

Rev. Page No.	No.
T2.4-11	-
T2.4-12	-
T2.4-13	6
T2.4-14	-
T2.4-15	-
T2.4-16	-
T2.4-17	-
T2.4-18 (Sh 1)	-
T2.4-18 (Sh 2)	-
T2.4-19	-
T2.4-20	-
T2.4-21	-
T2.4-22	-
T2.4-23	-
F2.4-1	-
F2.4-2	-
F2.4-3	-
F2.4-4	-
F2.4-5	-
F2.4-6	-
F2.4-7 (Sh 1)	-
F2.4-7 (Sh 2)	-
F2.4-8	-
F2.4-9	-
F2.4-10	-
F2.4-11	-
F2.4-12	-
F2.4-13	-
F2.4-14	-
F2.4-15	-
F2.4-16	-
F2.4-17	-
F2.4-18	-
F2.4-19	-
F2.4-20	6
F2.4-21	-
F2.4-22	-
F2.4-23	4
F2.4-24	-
F2.4-25	-
F2.4-26	-
F2.4-27	-
F2.4-28	-
F2.4-29	-
F2.4-30	-
F2.4-31	-
F2.4-32	-
2.5-1	-
2.5-2	-
2.5-3	-
2.5-4	-
2.5-5	-
2.5-6	-
2.5-7	-
2.5-8	-
2.5-9	-
2.5-10	-
2.5-11	-
2.5-12	-
2.5-13	-
2.5-14	-
2.5-15	-
2.5-16	-
2.5-17	-
2.5-18	-
2.5-19	-
2.5-20	-
2.5-21	-
2.5-22	-
2.5-23	-
2.5-24	-
2.5-25	-
2.5-26	-
2.5-27	-
2.5-28	-
2.5-29	-
2.5-30	-
2.5-31	-
2.5-32	-
2.5-33	-
2.5-34	-

Rev. Page No.	No.
2.5-35	-
2.5-36	-
2.5-37	-
2.5-38	-
2.5-39	-
2.5-40	-
2.5-41	-
2.5-42	-
2.5-43	-
2.5-44	-
2.5-45	-
2.5-46	-
2.5-47	-
2.5-48	-
2.5-49	-
2.5-50	-
2.5-51	-
2.5-52	-
2.5-53	-
2.5-54	-
2.5-55	-
2.5-56	-
2.5-57	-
2.5-58	-
2.5-59	-
2.5-60	-
2.5-61	-
2.5-62	-
2.5-63	-
2.5-64	-
2.5-65	-
2.5-66	-
2.5-67	-
2.5-68	-
2.5-69	-
2.5-70	-
2.5-71	-
2.5-72	-
2.5-73	-
2.5-74	-
2.5-75	-
2.5-76	-
2.5-77	-
2.5-78	-
2.5-79	-
2.5-80	-
2.5-81	-
2.5-82	-
2.5-83	-
2.5-84	-
2.5-85	-
2.5-86	-
2.5-87	-
2.5-88	-
2.5-89	-
2.5-90	-
2.5-91	-
2.5-92	-
2.5-93	-
2.5-94	-
2.5-95	-
2.5-96	-
2.5-97	-
2.5-98	-
2.5-99	-
2.5-100	-
2.5-101	-
2.5-102	-
2.5-103	-
2.5-104	-
2.5-105	-
2.5-106	-
2.5-107	-
2.5-108	-
2.5-109	-
2.5-110	-
2.5-111	-
2.5-112	-
2.5-113	-
2.5-114	-
2.5-115	-
2.5-116	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
2.5-117	-
2.5-118	-
2.5-119	-
2.5-120	-
2.5-121	-
2.5-122	-
2.5-123	-
2.5-124	-
2.5-125	-
2.5-126	-
2.5-127	-
2.5-128	-
2.5-129	-
2.5-130	-
2.5-131	-
2.5-132	-
2.5-133	-
2.5-134	-
2.5-135	6
2.5-135a	6
2.5-136	-
2.5-137	-
2.5-138	-
2.5-139	6
2.5-139a	6
2.5-140	-
2.5-141	-
2.5-142	-
2.5-143	-
2.5-144	-
2.5-145	-
2.5-146	-
2.5-147	-
2.5-148	-
2.5-149	-
2.5-150	-
2.5-151	-
2.5-152	-
2.5-153	-
2.5-154	-
2.5-155	-
2.5-156	-
2.5-157	-
2.5-158	-
2.5-159	-
2.5-160	-
2.5-161	-
2.5-162	-
2.5-163	-
2.5-164	-
2.5-165	-
2.5-166	-
2.5-167	-
2.5-168	-
2.5-169	-
2.5-170	-
T2.5-1 (Sh 1)	-
T2.5-1 (Sh 2)	-
T2.5-1 (Sh 3)	-
T2.5-2 (Sh 1)	-
T2.5-2 (Sh 2)	-
T2.5-2 (Sh 3)	-
T2.5-2 (Sh 4)	-
T2.5-2 (Sh 5)	-
T2.5-2 (Sh 6)	-
T2.5-2 (Sh 7)	-
T2.5-2 (Sh 8)	-
T2.5-2 (Sh 9)	-
T2.5-2 (Sh 10)	-
T2.5-2 (Sh 11)	-
T2.5-3 (Sh 1)	-
T2.5-3 (Sh 2)	-
T2.5-4	-
T2.5-5 (Sh 1)	-
T2.5-5 (Sh 2)	-
T2.5-5 (Sh 3)	-
T2.5-5 (Sh 4)	-
T2.5-5 (Sh 5)	-
T2.5-5 (Sh 6)	-
T2.5-5 (Sh 7)	-
T2.5-5 (Sh 8)	-
T2.5-5 (Sh 9)	-

Rev. Page No.	No.
T2.5-5 (Sh 10)	-
T2.5-5 (Sh 11)	-
T2.5-5 (Sh 12)	-
T2.5-5 (Sh 13)	-
T2.5-5 (Sh 14)	-
T2.5-5 (Sh 15)	-
T2.5-5 (Sh 16)	-
T2.5-5 (Sh 17)	-
T2.5-5 (Sh 18)	-
T2.5-5 (Sh 19)	-
T2.5-6 (Sh 1)	-
T2.5-6 (Sh 2)	-
T2.5-7	-
T2.5-8	-
T2.5-9	-
T2.5-10 (Sh 1)	-
T2.5-10 (Sh 2)	-
T2.5-10 (Sh 3)	-
T2.5-10 (Sh 4)	-
T2.5-11 (Sh 1)	-
T2.5-11 (Sh 2)	-
T2.5-11 (Sh 3)	-
T2.5-11 (Sh 4)	-
T2.5-11 (Sh 5)	-
T2.5-12 (Sh 1)	-
T2.5-12 (Sh 2)	-
T2.5-13	-
T2.5-14	-
T2.5-15 (Sh 1)	-
T2.5-15 (Sh 2)	-
T2.5-16	-
T2.5-17	-
T2.5-18 (Sh 1)	-
T2.5-18 (Sh 2)	-
T2.5-18 (Sh 3)	-
T2.5-18 (Sh 4)	-
T2.5-18 (Sh 5)	-
T2.5-19	-
T2.5-20 (Sh 1)	-
T2.5-20 (Sh 2)	-
T2.5-21	-
F2.5-1	-
F2.5-2	-
F2.5-3	-
F2.5-4	-
F2.5-5	-
F2.5-6	-
F2.5-7	-
F2.5-8	-
F2.5-9	-
F2.5-10	-
F2.5-11	-
F2.5-12	-
F2.5-13	-
F2.5-14	-
F2.5-15	-
F2.5-16	-
F2.5-17	-
F2.5-18	-
F2.5-19	-
F2.5-20	-
F2.5-21	-
F2.5-22	-
F2.5-23	-
F2.5-24	-
F2.5-25	-
F2.5-26	-
F2.5-27	-
F2.5-28	-
F2.5-29	-
F2.5-30	-
F2.5-31	-
F2.5-32	-
F2.5-33	-
F2.5-34	-
F2.5-35	-
F2.5-36	-
F2.5-37	-
F2.5-38	-
F2.5-39	-
F2.5-40	-
F2.5-41	-

Rev. Page No.	No.
F2.5-42	-
F2.5-43	-
F2.5-44	-
F2.5-45	-
F2.5-46	-
F2.5-47	-
F2.5-48	-
F2.5-49 (Sh 1)	-
F2.5-49 (Sh 2)	-
F2.5-50 (Sh 1)	-
F2.5-50 (Sh 2)	-
F2.5-51	-
F2.5-52	-
F2.5-53	-
F2.5-54	-
F2.5-55	-
F2.5-56	-
F2.5-57	-
F2.5-58	-
F2.5-59	-
F2.5-60	-
F2.5-61	-
F2.5-62	-
F2.5-63	-
F2.5-64	-
F2.5-65	-
F2.5-66	-
F2.5-67	-
F2.5-68	-
F2.5-69	-
F2.5-70	-
F2.5-71	-

VOLUME 3

Appendix 2A

All Sheets	*

Appendix 2B

All Sheets	*

Appendix 2C

All Sheets	*

Appendix 2D

All Sheets	*

Appendix 2E

All Sheets	*

Appendix 2F

All Sheets	*

VOLUME 4

Appendix 2G

All Sheets	*

Appendix 2H

All Sheets	*

Appendix 2I

All Sheets	*

Appendix 2J

All Sheets	*

Appendix 2K

All Sheets	*

Appendix 2L

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
All Sheets	*
Appendix 2M	
All Sheets	*
Appendix 2N	
All Sheets	*
Appendix 2O	
All Sheets	*
Appendix 2P	
All Sheets	*
VOLUME 5	
Chapter 3	
3.0-1	-
3.0-2	-
3.0-3	-
3.0-4	6
3.0-5	-
3.0-6	-
3.0-7	6
3.0-8	-
3.0-9	6
3.0-10	3
3.0-11	2
3.0-12	-
3.0-13	-
3.0-14	-
3.0-15	-
3.0-16	-
3.0-17	-
3.0-18	-
3.0-19	7
3.0-20	3
3.0-21	-
3.0-22	-
3.0-23	-
3.0-24	-
3.0-25	-
3.0-26	-
3.0-27	-
3.0-28	-
3.0-29	-
3.1-1	-
3.1-2	-
3.1-3	-
3.1-4	-
3.1-5	-
3.1-6	-
3.1-7	-
3.1-8	-
3.1-9	-
3.1-10	-
3.1-11	-
3.1-12	5
3.1-13	5
3.1-14	-
3.1-15	-
3.1-16	-
3.1-17	-
3.1-18	-
3.1-19	-
3.1-20	-
3.1-21	-
3.1-22	-
3.1-23	-
3.1-24	-
3.1-25	-
3.1-26	-
3.1-27	-
3.1-28	-
3.1-29	-

Rev. Page No.	No.
3.1-30	-
3.1-31	5
3.1-32	5
3.1-33	-
3.1-34	-
3.1-35	-
3.2-1	-
3.2-2	-
3.2-3	-
3.2-4	-
T3.2-1 (Sh 1)	6
T3.2-1 (Sh 2)	-
T3.2-1 (Sh 3)	-
T3.2-1 (Sh 4)	-
T3.2-1 (Sh 5)	-
T3.2-1 (Sh 6)	-
T3.2-2 (Sh 1)	-
T3.2-2 (Sh 2)	2
T3.2-2 (Sh 3)	-
T3.2-2 (Sh 4)	-
T3.2-2 (Sh 5)	-
T3.2-2 (Sh 6)	3
T3.2-2 (Sh 7)	-
T3.2-2 (Sh 8)	-
T3.2-2 (Sh 9)	-
T3.2-2 (Sh 10)	-
T3.2-2 (Sh 11)	-
T3.2-2 (Sh 12)	1
T3.2-2 (Sh 13)	1
T3.2-2 (Sh 14)	1
T3.2-2 (Sh 15)	1
T3.2-2 (Sh 16)	1
T3.2-2 (Sh 17)	1
T3.2-2 (Sh 18)	1
T3.2-2 (Sh 19)	1
T3.2-2 (Sh 20)	1
T3.2-2 (Sh 21)	-
T3.2-2 (Sh 22)	-
T3.2-2 (Sh 23)	-
T3.2-2 (Sh 24)	-
T3.2-2 (Sh 25)	7
T3.2-2 (Sh 25a)	7
T3.2-2 (Sh 26)	4
T3.2-2 (Sh 27)	-
T3.2-2 (Sh 28)	-
T3.2-2 (Sh 29)	-
T3.2-2 (Sh 30)	-
T3.2-2 (Sh 31)	-
T3.2-2 (Sh 32)	6
T3.2-2 (Sh 33)	6
T3.2-2 (Sh 34)	6
T3.2-2 (Sh 35)	-
T3.2-2 (Sh 36)	-
T3.2-2 (Sh 37)	-
T3.2-2 (Sh 38)	-
T3.2-2 (Sh 39)	-
T3.2-2 (Sh 40)	-
T3.2-2 (Sh 41)	-
T3.2-2 (Sh 42)	-
T3.2-2 (Sh 43)	-
T3.2-2 (Sh 44)	-
T3.2-2 (Sh 45)	6
T3.2-2 (Sh 46)	3
T3.2-3	-
T3.2-4	4
3.3-1	-
3.3-2	3
3.3-2a	2
3.3-3	3
3.3-3a	2
3.3-4	-
3.3-5	-
3.3-6	-
T3.3-1	-
T3.3-2	-
T3.3-3	-
T3.3-4 (Sh 1)	-
T3.3-4 (Sh 2)	-
F3.3-1	-
F3.3-2	-

Rev. Page No.	No.
3.4-1	-
3.4-2	-
F3.4-1	-
3.5-1	-
3.5-2	-
3.5-3	6
3.5-4	-
3.5-5	-
3.5-6	-
3.5-7	6
3.5-7a	6
3.5-8	-
3.5-9	-
3.5-10	-
3.5-11	-
3.5-12	-
3.5-13	-
3.5-14	-
3.5-15	-
3.5-16	-
3.5-17	-
3.5-18	-
3.5-19	2
3.5-19a	2
3.5-20	-
3.5-21	-
3.5-22	-
3.5-23	-
3.5-24	-
3.5-25	-
3.5-26	-
3.5-27	-
3.5-28	-
3.5-29	-
3.5-30	-
3.5-31	-
3.5-32	-
3.5-33	-
T3.5-1	-
T3.5-2	6
T3.5-3	-
T3.5-4	-
T3.5-5	-
T3.5-6	-
T3.5-7	-
T3.5-8	-
T3.5-9	-
T3.5-10	-
T3.5-11	-
T3.5-12	2
T3.5-13	-
F3.5-1	-
F3.5-2	-
F3.5-3	-
F3.5-4	-
F3.5-5	-
3.6(B)-1	6
3.6(B)-1a	6
3.6(B)-2	-
3.6(B)-3	6
3.6(B)-3a	6
3.6(B)-4	-
3.6(B)-5	7
3.6(B)-6	-
3.6(B)-7	6
3.6(B)-8	6
3.6(B)-9	6
3.6(B)-10	6
3.6(B)-11	6
3.6(B)-11a	6
3.6(B)-12	7
3.6(B)-13	-
3.6(B)-14	-
3.6(B)-15	-
3.6(B)-16	-
3.6(B)-17	-
3.6(B)-18	-
3.6(B)-19	-
3.6(B)-20	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
3.6(B)-21	7
T3.6(B)-1	7
T3.6(B)-2 (Sh 1)	7
T3.6(B)-2 (Sh 2)	7
T3.6(B)-2 (Sh 3)	7
T3.6(B)-2 (Sh 4)	7
T3.6(B)-2 (Sh 5)	7
T3.6(B)-2 (Sh 6)	7
T3.6(B)-2 (Sh 7)	7
T3.6(B)-2 (Sh 8)	7
T3.6(B)-2 (Sh 9)	7
T3.6(B)-2 (Sh 10)	7
T3.6(B)-2 (Sh 11)	7
T3.6(B)-2 (Sh 12)	7
T3.6(B)-2 (Sh 13)	7
F3.6(B)-1	7
F3.6(B)-2	7
F3.6(B)-3	-
F3.6(B)-4	-
F3.6(B)-5	2
F3.6(B)-6	-
F3.6(B)-7	2
F3.6(B)-8	2
F3.6(B)-9	2
F3.6(B)-10	2
F3.6(B)-11	2
F3.6(B)-12	2
F3.6(B)-13 (Sh 1)	-
F3.6(B)-13 (Sh 2)	-
F3.6(B)-14	-
F3.6(B)-15	7
F3.6(B)-16	7
F3.6(B)-17	7
F3.6(B)-18	7
F3.6(B)-19	-
F3.6(B)-20	7
F3.6(B)-21	7
F3.6(B)-22	7
F3.6(B)-23	7
F3.6(B)-24	7
F3.6(B)-25	-
F3.6(B)-26	7
F3.6(B)-27	7
F3.6(B)-28	7
F3.6(B)-29	-
F3.6(B)-30	7
F3.6(B)-31	7
F3.6(B)-32	7
F3.6(B)-33	7
F3.6(B)-34	7
3.6(N)-1	-
3.6(N)-2	-
3.6(N)-3	-
3.6(N)-4	-
3.6(N)-5	-
3.6(N)-6	-
3.6(N)-7	-
3.6(N)-8	-
T3.6(N)-1	-
F3.6(N)-1	-
F3.6(N)-1	-
F3.6(N)-2	-
3.7(B)-1	-
3.7(B)-2	-
3.7(B)-3	-
3.7(B)-4	-
3.7(B)-5	-
3.7(B)-6	-
3.7(B)-7	-
3.7(B)-8	-
3.7(B)-9	-
3.7(B)-10	-
3.7(B)-11	-
3.7(B)-12	-
3.7(B)-13	-
3.7(B)-14	-
3.7(B)-15	-
3.7(B)-16	-
3.7(B)-17	-
3.7(B)-18	-
3.7(B)-19	-
3.7(B)-20	-
3.7(B)-21	6
3.7(B)-22	6
3.7(B)-23	6
3.7(B)-24	6
3.7(B)-25	6
T3.7(B)-1	-
T3.7(B)-2	-
T3.7(B)-3	-
T3.7(B)-4	-
T3.7(B)-5	-
T3.7(B)-6	-
T3.7(B)-7	-
T3.7(B)-8 (Sh 1)	-
T3.7(B)-8 (Sh 2)	-
T3.7(B)-8 (Sh 3)	-
T3.7(B)-9	-
T3.7(B)-10 (Sh 1)	-
T3.7(B)-10 (Sh 2)	-
T3.7(B)-11 (Sh 1)	-
T3.7(B)-11 (Sh 2)	-
T3.7(B)-12 (Sh 1)	-
T3.7(B)-12 (Sh 2)	-
T3.7(B)-13 (Sh 1)	-
T3.7(B)-13 (Sh 2)	-
T3.7(B)-14	-
T3.7(B)-15	-
T3.7(B)-16 (Sh 1)	-
T3.7(B)-16 (Sh 2)	-
T3.7(B)-17	-
T3.7(B)-18 (Sh 1)	-
T3.7(B)-18 (Sh 2)	-
T3.7(B)-18 (Sh 3)	-
T3.7(B)-19 (Sh 1)	-
T3.7(B)-19 (Sh 2)	-
T3.7(B)-19 (Sh 3)	-
T3.7(B)-19 (Sh 4)	-
T3.7(B)-19 (Sh 5)	-
T3.7(B)-20 (Sh 1)	-
T3.7(B)-20 (Sh 2)	-
T3.7(B)-21	-
T3.7(B)-22 (Sh 1)	-
T3.7(B)-22 (Sh 2)	-
T3.7(B)-23	-
T3.7(B)-24	6
F3.7(B)-1	-
F3.7(B)-2	-
F3.7(B)-3	-
F3.7(B)-4	-
F3.7(B)-5	-
F3.7(B)-6	-
F3.7(B)-7	-
F3.7(B)-8	-
F3.7(B)-9	-
F3.7(B)-10	-
F3.7(B)-11	-
F3.7(B)-12	-
F3.7(B)-13	-
F3.7(B)-14	-
F3.7(B)-15	-
F3.7(B)-16	-
F3.7(B)-17	-
F3.7(B)-18	-
F3.7(B)-19	-
F3.7(B)-20	-
F3.7(B)-21	-
F3.7(B)-22	-
F3.7(B)-23	-
F3.7(B)-24	-
F3.7(B)-25	-
F3.7(B)-26 (Sh 1)	-
F3.7(B)-26 (Sh 2)	-
F3.7(B)-27 (Sh 1)	-
F3.7(B)-27 (Sh 2)	-
F3.7(B)-27 (Sh 3)	-
F3.7(B)-27 (Sh 4)	-
F3.7(B)-28 (Sh 1)	-
F3.7(B)-28 (Sh 2)	-
F3.7(B)-28 (Sh 3)	-
F3.7(B)-28 (Sh 4)	-
F3.7(B)-28 (Sh 5)	-
F3.7(B)-28 (Sh 6)	-
F3.7(B)-28 (Sh 7)	-
F3.7(B)-28 (Sh 8)	-
F3.7(B)-29 (Sh 1)	-
F3.7(B)-29 (Sh 2)	-
F3.7(B)-29 (Sh 3)	-
F3.7(B)-29 (Sh 4)	-
F3.7(B)-30	-
F3.7(B)-31	-
F3.7(B)-32	6
F3.7(B)-33	-
F3.7(B)-34	-
F3.7(B)-35	-
F3.7(B)-36	-
F3.7(B)-37	-
F3.7(B)-38	-
3.7(N)-1	-
3.7(N)-2	-
3.7(N)-3	-
3.7(N)-4	-
3.7(N)-5	-
3.7(N)-6	-
3.7(N)-7	-
3.7(N)-8	-
3.7(N)-9	-
3.7(N)-10	-
3.7(N)-11	-
3.7(N)-12	-
3.7(N)-13	-
3.7(N)-14	-
3.7(N)-15	-
3.7(N)-16	-
3.7(N)-17	-
3.7(N)-18	-
3.7(N)-19	-
3.7(N)-20	-
3.7(N)-21	-
T3.7(N)-1	-
F3.7(N)-1	-
3.8-1	-
3.8-2	-
3.8-3	-
3.8-4	-
3.8-5	-
3.8-6	-
3.8-7	-
3.8-8	-
3.8-9	-
3.8-10	-
3.8-11	-
3.8-12	-
3.8-13	-
3.8-14	-
3.8-15	-
3.8-16	-
3.8-17	-
3.8-18	-
3.8-19	-
3.8-20	-
3.8-21	-
3.8-22	-
3.8-23	-
3.8-24	-
3.8-25	-
3.8-26	-
3.8-27	-
3.8-28	-
3.8-29	-
3.8-30	-
3.8-31	-
3.8-32	-
3.8-33	-
3.8-34	-
3.8-35	-
3.8-36	-
3.8-37	-
3.8-38	-
3.8-39	-
3.8-40	-
3.8-41	-
3.8-42	-
3.8-43	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
3.8-44	3
3.8-45	3
3.8-46	5
3.8-47	3
3.8-48	6
3.8-49	5
3.8-49a	5
3.8-50	3
3.8-51	3
3.8-52	3
3.8-53	3
3.8-54	3
3.8-55	5
3.8-55a	5
3.8-56	3
3.8-57	3
3.8-58	3
3.8-59	3
3.8-60	3
3.8-61	3
3.8-62	3
3.8-63	3
3.8-64	3
3.8-65	3
3.8-66	3
3.8-67	3
3.8-68	3
3.8-69	3
3.8-70	3
3.8-71	3
3.8-72	3
3.8-73	3
3.8-74	3
3.8-75	3
3.8-76	3
3.8-77	3
3.8-78	3
3.8-79	3
3.8-80	3
3.8-81	3
3.8-82	5
3.8-83	3
3.8-84	3
3.8-85	6
3.8-85a	6
3.8-86	3
3.8-87	3
3.8-88	3
3.8-89	3
3.8-90	3
3.8-91	3
3.8-92	3
3.8-93	3
3.8-94	3
3.8-95	3
3.8-96	3
3.8-97	3
3.8-98	3
3.8-99	3
3.8-100	3
3.8-101	3
3.8-102	3
3.8-103	3
3.8-104	3
3.8-105	3
3.8-106	3
3.8-107	3
3.8-108	6
3.8-109	6
3.8-109a	6
3.8-110	3
3.8-111	3
3.8-112	3
3.8-113	3
3.8-114	3
3.8-115	3
3.8-116	3
3.8-117	3
3.8-118	3
3.8-119	3
3.8-120	3

Page No.	Rev. No.
3.8-121	3
3.8-122	3
3.8-123	6
3.8-123a	6
3.8-124	3
3.8-125	3
3.8-126	3
3.8-127	3
3.8-128	3
3.8-129	3
3.8-130	3
3.8-131	3
3.8-132	3
3.8-133	5
3.8-134	3
3.8-135	3
3.8-136	3
3.8-137	3
3.8-138	3
3.8-139	3
T3.8-1 (Sh 1)	-
T3.8-1 (Sh 2)	-
T3.8-2	-
T3.8-3	-
T3.8-4	-
T3.8-5	-
T3.8-6	-
T3.8-7	-
T3.8-8	-
T3.8-9	-
T3.8-10	-
T3.8-11 (Sh 1)	-
T3.8-11 (Sh 2)	-
T3.8-12	-
T3.8-13 (Sh 1)	-
T3.8-13 (Sh 2)	-
T3.8-13 (Sh 2a)	3
T3.8-14	-
T3.8-15 (Sh 1)	-
T3.8-15 (Sh 2)	6
T3.8-16	-
T3.8-17	-
F3.8-1 (Sh 1)	-
F3.8-1 (Sh 2)	-
F3.8-2	-
F3.8-3 (Sh 1)	-
F3.8-3 (Sh 2)	-
F3.8-4 (Sh 1)	-
F3.8-4 (Sh 2)	-
F3.8-5 (Sh 1)	-
F3.8-5 (Sh 2)	-
F3.8-6	-
F3.8-7	-
F3.8-8	-
F3.8-9	-
F3.8-10	-
F3.8-11	-
F3.8-12	-
F3.8-13	-
F3.8-14	-
F3.8-15	-
F3.8-16	-
F3.8-17	-
F3.8-18	-
F3.8-19	-
F3.8-20	-
F3.8-21	-
F3.8-22	-
F3.8-23	-
F3.8-24	-
F3.8-25	-
F3.8-26	3
F3.8-27	-
F3.8-28	-
F3.8-29	-
F3.8-30	-
F3.8-31	-
F3.8-32	-
F3.8-33	-
F3.8-34	-
F3.8-35	-
F3.8-36	-

Page No.	Rev. No.
F3.8-37	6
3.9(B)-1	-
3.9(B)-2	-
3.9(B)-3	-
3.9(B)-4	-
3.9(B)-5	-
3.9(B)-6	-
3.9(B)-7	-
3.9(B)-8	-
3.9(B)-9	-
3.9(B)-10	-
3.9(B)-11	-
3.9(B)-12	-
3.9(B)-13	-
3.9(B)-14	-
3.9(B)-15	-
3.9(B)-16	7
3.9(B)-17	7
3.9(B)-18	7
3.9(B)-19	-
3.9(B)-20	-
3.9(B)-21	-
3.9(B)-22	-
3.9(B)-23	-
3.9(B)-24	-
3.9(B)-25	-
3.9(B)-26	-
3.9(B)-27	7
3.9(B)-28	-
3.9(B)-29	7
3.9(B)-30	7
3.9(B)-31	-
3.9(B)-32	-
3.9(B)-33	-
3.9(B)-34	-
3.9(B)-35	-
3.9(B)-36	-
3.9(B)-37	-
3.9(B)-38	2
3.9(B)-39	7
3.9(B)-40	7
3.9(B)-41	7
3.9(B)-42	7
T3.9(B)-1 (Sh 1)	-
T3.9(B)-1 (Sh 2)	6
T3.9(B)-2 (Sh 1)	-
T3.9(B)-2 (Sh 2)	-
T3.9(B)-3	-
T3.9(B)-4	-
T3.9(B)-5	-
T3.9(B)-6	-
T3.9(B)-7	-
T3.9(B)-8	7
T3.9(B)-9 (Sh 1)	7
T3.9(B)-9 (Sh 2)	-
T3.9(B)-10	-
T3.9(B)-11 (Sh 1)	7
T3.9(B)-11 (Sh 2)	7
T3.9(B)-12	-
T3.9(B)-13	-
T3.9(B)-14	-
T3.9(B)-15	7
T3.9(B)-16	7
T3.9(B)-17 (Sh 1)	7
T3.9(B)-17 (Sh 2)	7
T3.9(B)-18	7
T3.9(B)-19	7
T3.9(B)-20	-
T3.9(B)-21	7
T3.9(B)-22 (Sh 1)	7
T3.9(B)-22 (Sh 2)	1
T3.9(B)-22 (Sh 3)	1
T3.9(B)-22 (Sh 4)	1
T3.9(B)-22 (Sh 5)	1
T3.9(B)-22 (Sh 6)	1
T3.9(B)-22 (Sh 7)	1
T3.9(B)-22 (Sh 8)	1
T3.9(B)-22 (Sh 9)	1
T3.9(B)-22 (Sh 10)	1
T3.9(B)-23 (Sh 1)	-
T3.9(B)-23 (Sh 2)	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
T3.9(B)-24	3
T3.9(B)-25	3
T3.9(B)-26	-
T3.9(B)-27 (Sh 1)	6
T3.9(B)-27 (Sh 2)	6
T3.9(B)-27 (Sh 3)	6
T3.9(B)-27 (Sh 4)	6
T3.9(B)-27 (Sh 5)	6
T3.9(B)-27 (Sh 6)	6
T3.9(B)-27 (Sh 7)	6
T3.9(B)-27 (Sh 8)	6
3.9(N)-1	-
3.9(N)-2	-
3.9(N)-3	-
3.9(N)-4	-
3.9(N)-5	-
3.9(N)-6	-
3.9(N)-7	-
3.9(N)-8	7
3.9(N)-9	-
3.9(N)-10	-
3.9(N)-11	-
3.9(N)-12	-
3.9(N)-13	-
3.9(N)-14	-
3.9(N)-15	-
3.9(N)-16	-
3.9(N)-17	-
3.9(N)-18	-
3.9(N)-19	-
3.9(N)-20	-
3.9(N)-21	-
3.9(N)-22	-
3.9(N)-23	-
3.9(N)-24	-
3.9(N)-25	-
3.9(N)-26	-
3.9(N)-27	-
3.9(N)-28	-
3.9(N)-29	-
3.9(N)-30	-
3.9(N)-31	-
3.9(N)-32	-
3.9(N)-33	-
3.9(N)-34	7
3.9(N)-35	-
3.9(N)-36	-
3.9(N)-37	-
3.9(N)-38	-
3.9(N)-39	-
3.9(N)-40	-
3.9(N)-41	-
3.9(N)-42	-
3.9(N)-43	-
3.9(N)-44	-
3.9(N)-45	-
3.9(N)-46	7
3.9(N)-46a	7
3.9(N)-47	-
3.9(N)-48	-
3.9(N)-49	7
3.9(N)-50	-
3.9(N)-51	-
3.9(N)-52	-
3.9(N)-53	-
3.9(N)-54	-
3.9(N)-55	-
3.9(N)-56	-
3.9(N)-57	-
3.9(N)-58	-
3.9(N)-59	-
3.9(N)-60	-
3.9(N)-61	-
3.9(N)-62	-
3.9(N)-63	-
3.9(N)-64	-
3.9(N)-65	-
3.9(N)-66	-
3.9(N)-67	-
3.9(N)-68	-
3.9(N)-69	-
3.9(N)-70	-

Page No.	Rev. No.
3.9(N)-71	-
3.9(N)-72	-
3.9(N)-73	-
3.9(N)-74	-
3.9(N)-75	-
3.9(N)-76	-
3.9(N)-77	7
3.9(N)-78	7
3.9(N)-79	7
T3.9(N)-1 (Sh 1)	-
T3.9(N)-1 (Sh 2)	-
T3.9(N)-1 (Sh 3)	-
T3.9(N)-2	-
T3.9(N)-3	-
T3.9(N)-4	-
T3.9(N)-5	-
T3.9(N)-6	-
T3.9(N)-7	-
T3.9(N)-8	-
T3.9(N)-9	-
T3.9(N)-10	7
T3.9(N)-11 (Sh 1)	5
T3.9(N)-11 (Sh 2)	5
T3.9(N)-11 (Sh 3)	5
T3.9(N)-12	-
F3.9(N)-1	-
F3.9(N)-2	-
F3.9(N)-3	-
F3.9(N)-4	-
F3.9(N)-5	-
F3.9(N)-6	-
F3.9(N)-7	-
F3.9(N)-8	-
F3.9(N)-9	-
3.10(B)-1	-
3.10(B)-2	-
3.10(B)-3	-
3.10(B)-4	6
T3.10(B)-1 (Sh 1)	-
T3.10(B)-1 (Sh 2)	-
T3.10(B)-1 (Sh 3)	-
T3.10(B)-1 (Sh 4)	-
3.10(N)-1	-
3.10(N)-2	-
3.10(N)-3	-
3.10(N)-4	-
T3.10(N)-1	-
3.11-1	-
3.11-2	-
3.11-3	6
3.11-3a	6
3.11-4	-
3.11-5	-
3.11-6	-
F3.11-1 (Sh 1)	4
F3.11-1 (Sh 2)	-
F3.11-1 (Sh 3)	-
F3.11-1 (Sh 4)	5
F3.11-1 (Sh 5)	-

VOLUME 6

Appendix 3A

Page No.	Rev. No.
All Sheets	

Appendix 3B

Page No.	Rev. No.
Delete Sheet	*

Appendix 3C

Page No.	Rev. No.
All Sheets	*

Appendix 3D

Page No.	Rev. No.
All Sheets	*

Appendix 3E

Page No.	Rev. No.
All Sheets	*

Appendix 3F

Page No.	Rev. No.
All Sheets	*

Appendix 3G

Page No.	Rev. No.
All Sheets	*

Appendix 3H

Page No.	Rev. No.
Delete Sheet	*

Appendix 3I

Page No.	Rev. No.
All Sheets	*

Chapter 4

Page No.	Rev. No.
4.0-1	6
4.0-2	4
4.0-3	4
4.0-4	4
4.0-5	4
4.0-6	7
4.0-7	7
4.0-8	7
4.0-9	7
4.0-10	7
4.0-11	7
4.1-1	7
4.1-2	7
4.1-3	7
T4.1-1 (Sh 1)	4
T4.1-1 (Sh 2)	4
T4.1-1 (Sh 3)	4
T4.1-2 (Sh 1)	7
T4.1-2 (Sh 2)	4
T4.1-3	-
4.2-1	4
4.2-2	7
4.2-3	7
4.2-4	7
4.2-5	4
4.2-6	4
4.2-7	6
4.2-7a	6
4.2-8	4
4.2-9	4
4.2-10	6
4.2-11	4
4.2-12	4
4.2-13	4
4.2-14	4
4.2-15	4
4.2-16	4
4.2-17	4
4.2-18	4
4.2-19	4
4.2-20	7
4.2-21	7
4.2-22	7
4.2-23	7
4.2-23a	7
4.2-23b	7
4.2-24	4
4.2-25	4
4.2-26	4
4.2-27	4
4.2-28	4
4.2-29	7
4.2-30	4
4.2-31	4
4.2-32	4
4.2-33	4
4.2-34	4
4.2-35	4
4.2-36	4
4.2-37	4

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
4.2-38	4
4.2-39	7
4.2-40	7
F4.2-1	-
F4.2-2	7
F4.2-2A	4
F4.2-2B	7
F4.2-3	7
F4.2-3A	4
F4.2-3B	7
F4.2-4	-
F4.2-5	-
F4.2-6	7
F4.2-6A	4
F4.2-6B	4
F4.2-7	-
F4.2-8	-
F4.2-9	7
F4.2-9A	7
F4.2-9B	7
F4.2-10	-
F4.2-10A	4
F4.2-10B	7
F4.2-11	7
F4.2-12	7
F4.2-13	-
F4.2-14	7
F4.2-14A	4
F4.2-14B	4
F4.2-15	-
4.3-1	4
4.3-2	4
4.3-3	4
4.3-4	4
4.3-5	4
4.3-6	6
4.3-7	4
4.3-8	4
4.3-9	4
4.3-10	4
4.3-11	4
4.3-12	4
4.3-13	4
4.3-14	4
4.3-15	4
4.3-16	4
4.3-17	4
4.3-18	4
4.3-19	4
4.3-20	4
4.3-21	4
4.3-22	4
4.3-23	4
4.3-24	4
4.3-25	4
4.3-26	7
4.3-27	4
4.3-28	4
4.3-29	4
4.3-30	4
4.3-31	4
4.3-32	4
4.3-33	4
4.3-34	4
4.3-35	4
4.3-36	4
4.3-37	4
4.3-38	7
4.3-39	7
4.3-40	7
4.3-41	7
T4.3-1 (Sh 1)	4
T4.3-1 (Sh 2)	4
T4.3-2 (Sh 1)	7
T4.3-2 (Sh 2)	4
T4.3-3	7
T4.3-4	4
T4.3-5	4
T4.3-6	4
T4.3-7	4
T4.3-8	4

Rev. Page No.	No.
T4.3-9	4
T4.3-10	4
F4.3-1	4
F4.3-2	4
F4.3-3	4
F4.3-4	4
F4.3-5	4
F4.3-6	4
F4.3-7	4
F4.3-8	4
F4.3-9	4
F4.3-10	4
F4.3-11	4
F4.3-12	4
F4.3-13	4
F4.3-14	4
F4.3-15	4
F4.3-16	4
F4.3-17	-
F4.3-18	4
F4.3-19	4
F4.3-20	4
F4.3-21	4
F4.3-22	4
F4.3-23	4
F4.3-24	4
F4.3-25	4
F4.3-26	4
F4.3-27	4
F4.3-28	4
F4.3-29	4
F4.3-30	4
F4.3-31	4
F4.3-32	4
F4.3-33	4
F4.3-34	4
F4.3-35	4
F4.3-36	-
F4.3-37	-
F4.3-38	-
F4.3-39	4
F4.3-40	-
F4.3-41	-
F4.3-42	-
F4.3-43	4
F4.3-44	4
F4.3-45	4
4.4-1	7
4.4-2	7
4.4-3	7
4.4-4	7
4.4-5	7
4.4-6	7
4.4-7	7
4.4-8	7
4.4-9	7
4.4-10	7
4.4-11	7
4.4-12	7
4.4-13	7
4.4-14	7
4.4-15	7
4.4-16	7
4.4-17	7
4.4-18	7
4.4-19	7
4.4-20	7
4.4-21	7
4.4-22	7
4.4-23	7
4.4-24	7
4.4-25	7
4.4-25a	7
4.4-26	7
4.4-27	7
4.4-28	7
4.4-29	7
4.4-30	7
4.4-31	7
4.4-32	7
4.4-33	7

Rev. Page No.	No.
4.4-34	7
4.4-35	7
4.4-36	7
4.4-37	7
4.4-38	7
4.4-39	7
4.4-40	7
4.4-41	7
4.4-42	7
4.4-43	7
4.4-44	7
4.4-45	7
4.4-46	7
T4.4-1 (Sh 1)	7
T4.4-1 (Sh 2)	7
T4.4-2	7
F4.4-1	4
F4.4-2	4
F4.4-3	4
F4.4-4	4
F4.4-5	7
F4.4-6	4
F4.4-7	4
F4.4-8	4
F4.4-9	4
F4.4-10	4
F4.4-11	4
F4.4-12	4
4.5-1	-
4.5-2	-
4.5-3	-
4.6-1	-
4.6-2	-
Chapter 5	
5.0-1	1
5.0-2	1
5.0-3	-
5.0-4	2
5.0-5	1
5.0-6	6
5.0-7	-
5.0-8	-
5.0-9	-
5.1-1	-
5.1-2	-
5.1-3	-
5.1-4	-
5.1-5	-
5.1-6	-
T5.1-1 (Sh 1)	5
T5.1-1 (Sh 2)	4
F5.1-1	2
F5.1-2	7
F5.1-3 (Sh 1)	7
F5.1-3 (Sh 2)	6
F5.1-3 (Sh 3)	6
F5.1-3 (Sh 4)	6
F5.1-4	6
F5.1-5	2
Notes to F5.1-5	2
5.2-1	-
5.2-2	-
5.2-3	-
5.2-4	4
5.2-5	4
5.2-6	4
5.2-6a	4
5.2-7	-
5.2-8	-
5.2-9	-
5.2-10	-
5.2-11	-
5.2-12	7
5.2-13	7
5.2-14	6
5.2-14a	4
5.2-15	1

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
5.2-16	1
5.2-17	1
5.2-18	1
5.2-19	1
5.2-20	1
5.2-21	1
5.2-22	1
5.2-23	1
5.2-24	1
5.2-25	1
5.2-26	1
5.2-27	1
5.2-28	1
5.2-29	6
5.2-30	6
5.2-31	3
5.2-31a	3
5.2-32	1
5.2-33	1
5.2-34	6
5.2-34a	3
5.2-35	7
5.2-36	1
5.2-37	2
T5.2-1	-
T5.2-2 (Sh 1)	7
T5.2-2 (Sh 2)	7
T5.2-2 (Sh 3)	2
T5.2-3 (Sh 1)	-
T5.2-3 (Sh 2)	-
T5.2-4	-
T5.2-5	6
T5.2-6	2
T5.2-7	7
T5.2-8	-
F5.2-1	-
F5.2-2 (Sh 1)	3
F5.2-2 (Sh 2)	7
5.3-1	-
5.3-2	-
5.3-3	-
5.3-4	-
5.3-5	-
5.3-6	-
5.3-7	-
5.3-8	-
5.3-9	-
5.3-10	-
5.3-11	7
5.3-12	-
5.3-13	-
5.3-14	-
5.3-15	-
5.3-16	-
5.3-17	-
5.3-18	-
T5.3-1	-
T5.3-2 (Sh 1)	-
T5.3-2 (Sh 2)	-
T5.3-3	4
T5.3-4	-
T5.3-5 (Sh 1)	-
T5.3-5 (Sh 2)	-
T5.3-6	-
F5.3-1	-
5.4-1	4
5.4-2	-
5.4-3	-
5.4-4	5
5.4-5	-
5.4-6	-
5.4-7	-
5.4-8	-
5.4-9	-
5.4-10	-
5.4-11	-
5.4-12	-
5.4-13	-
5.4-14	-
5.4-15	-

Rev. Page No.	No.
5.4-16	-
5.4-17	-
5.4-18	-
5.4-19	6
5.4-20	6
5.4-21	-
5.4-22	2
5.4-23	2
5.4-24	6
5.4-25	7
5.4-26	7
5.4-27	2
5.4-28	7
5.4-29	7
5.4-29a	7
5.4-30	2
5.4-31	2
5.4-32	7
5.4-32a	7
5.4-33	2
5.4-34	7
5.4-35	4
5.4-36	1
5.4-37	1
5.4-38	7
5.4-39	7
5.4-39a	7
5.4-40	1
5.4-41	1
5.4-42	1
5.4-43	1
5.4-44	1
5.4-45	1
5.4-46	1
5.4-47	6
5.4-48	6
5.4-49	1
5.4-50	1
5.4-51	1
5.4-52	1
5.4-53	1
5.4-54	1
5.4-55	1
5.4-56	4
5.4-57	4
5.4-58	4
5.4-59	4
5.4-60	1
5.4-61	1
5.4-62	1
5.4-63	1
5.4-64	1
T5.4-1 (Sh 1)	-
T5.4-1 (Sh 2)	-
T5.4-2	-
T5.4-3	-
T5.4-4 (Sh 1)	-
T5.4-4 (Sh 2)	-
T5.4-5	-
T5.4-6	-
T5.4-7	-
T5.4-8	-
T5.4-9	-
T5.4-10	-
T5.4-11	-
T5.4-12	4
T5.4-13	4
T5.4-14	-
T5.4-15	-
T5.4-16	4
T5.4-17 (Sh 1)	7
T5.4-17 (Sh 2)	1
T5.4-17 (Sh 3)	-
T5.4-17 (Sh 4)	-
T5.4-17 (Sh 5)	-
T5.4-17 (Sh 6)	-
F5.4-1	-
F5.4-2	-
F5.4-3	-
F5.4-4	-
F5.4-5	-
F5.4-6	-

Rev. Page No.	No.
F5.4-7 (Sh 1)	6
F5.4-7 (Sh 2)	5
F5.4-8	-
F5.4-9	-
F5.4-10	7
F5.4-11	7
F5.4-12	-
Notes to F5.4-12 (Sh 1)	1
Notes to F5.4-12 (Sh 2)	-
F5.4-13	-
F5.4-14	-
F5.4-15	-
F5.4-16	-
F5.4-17	-
F5.4-18	-
F5.4-19	-

VOLUME 7

Chapter 6

Rev. Page No.	No.
6.0-1	-
6.0-2	5
6.0-3	1
6.0-4	1
6.0-5	7
6.0-6	-
6.0-7	-
6.0-8	4
6.0-9	4
6.0-10	4
6.0-11	4
6.0-12	4
6.0-13	7
6.0-14	7
6.0-15	6
6.0-16	4
6.0-17	4
6.0-18	4
6.0-19	4
6.0-20	4
6.0-21	4
6.1(B)-1	-
6.1(B)-2	7
6.1(B)-3	6
T6.1(B)-1 (Sh 1)	-
T6.1(B)-1 (Sh 2)	-
T6.1(B)-1 (Sh 3)	-
T6.1(B)-2	-
T6.1(B)-3	3
T6.1(B)-4	-
6.1(N)-1	-
6.1(N)-2	4
6.1(N)-3	-
T6.1(N)-1 (Sh 1)	-
T6.1(N)-1 (Sh 2)	-
T6.1(N)-2	-
6.2-1	-
6.2-2	-
6.2-3	-
6.2-4	-
6.2-5	-
6.2-6	-
6.2-7	-
6.2-8	-
6.2-9	7
6.2-10	-
6.2-11	-
6.2-12	6
6.2-12a	6
6.2-13	-
6.2-14	-
6.2-15	-
6.2-16	4
6.2-17	-
6.2-18	-
6.2-19	-
6.2-20	4
6.2-20a	4

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
6.2-21	4
6.2-22	4
6.2-22a	4
6.2-23	-
6.2-24	-
6.2-25	4
6.2-26	4
6.2-27	4
6.2-28	4
6.2-29	4
6.2-30	-
6.2-31	-
6.2-32	-
6.2-33	-
6.2-34	-
6.2-35	-
6.2-36	-
6.2-37	-
6.2-38	-
6.2-39	-
6.2-40	4
6.2-41	-
6.2-42	-
6.2-43	-
6.2-44	-
6.2-45	7
6.2-46	-
6.2-47	4
6.2-48	-
6.2-49	6
6.2-49a	7
6.2-50	-
6.2-51	-
6.2-52	-
6.2-53	-
6.2-54	-
6.2-55	-
6.2-56	-
6.2-57	7
6.2-58	-
6.2-59	-
6.2-60	2
6.2-61	-
6.2-62	-
6.2-63	5
6.2-64	-
6.2-65	-
6.2-66	-
6.2-67	-
6.2-68	5
6.2-69	5
6.2-69a	3
6.2-70	7
6.2-71	7
6.2-72	5
6.2-73	5
6.2-74	5
6.2-75	5
6.2-76	5
6.2-77	5
6.2-78	5
6.2-79	5
6.2-80	5
6.2-81	5
6.2-82	5
6.2-83	5
6.2-84	5
6.2-85	5
6.2-86	5
6.2-87	7
6.2-88	5
6.2-89	5
6.2-90	5
6.2-91	5
6.2-92	5
6.2-93	5
6.2-94	5
T6.2-1	6
T6.2-2 (Sh 1)	-
T6.2-2 (Sh 2)	-
T6.2-2 (Sh 3)	5
T6.2-3	-

Rev. Page No.	No.
T6.2-4	-
T6.2-5	-
T6.2-6	-
T6.2-7	-
T6.2-8	-
T6.2-9	-
T6.2-10	-
T6.2-11	-
T6.2-12	-
T6.2-13	4
T6.2-14	4
T6.2-15	4
T6.2-16	4
T6.2-17a	4
T6.2-17b	4
T6.2-18	4
T6.2-19	4
T6.2-20	4
T6.2-21a	4
T6.2-21b	4
T6.2-22	4
T6.2-23a	4
T6.2-23b	4
T6.2-24	4
T6.2-25	4
T6.2-26	4
T6.2-27	4
T6.2-28	4
T6.2-29	4
T6.2-30	-
T6.2-31 (Sh 1)	-
T6.2-31 (Sh 2)	-
T6.2-31 (Sh 3)	-
T6.2-32 (Sh 1)	-
T6.2-32 (Sh 2)	-
T6.2-32 (Sh 3)	-
T6.2-32 (Sh 4)	-
T6.2-32 (Sh 5)	-
T6.2-32 (Sh 6)	-
T6.2-33	-
T6.2-34	-
T6.2-35	-
T6.2-36	-
T6.2-37	-
T6.2-38	-
T6.2-39	-
T6.2-40	-
T6.2-41	-
T6.2-42 (Sh 1)	-
T6.2-42 (Sh 2)	-
T6.2-43	-
T6.2-44	-
T6.2-45	-
T6.2-46	-
T6.2-47 (Sh 1)	-
T6.2-47 (Sh 2)	-
T6.2-48	-
T6.2-49	-
T6.2-50	-
T6.2-51	-
T6.2-52	-
T6.2-53	-
T6.2-54	-
T6.2-55	-
T6.2-56	-
T6.2-57 (Sh 1)	-
T6.2-57 (Sh 2)	-
T6.2-58 (Sh 1)	-
T6.2-58 (Sh 2)	-
T6.2-59 (Sh 1)	-
T6.2-59 (Sh 2)	-
T6.2-60	-
T6.2-61	-
T6.2-62	-
T6.2-63	-
T6.2-64	-
T6.2-65	-
T6.2-66 (Sh 1)	-
T6.2-66 (Sh 2)	4
T6.2-67	-
T6.2-68 (Sh 1)	-

Rev. Page No.	No.
T6.2-68 (Sh 2)	4
T6.2-69	-
T6.2-69a	4
T6.2-69b	4
T6.2-70 (Sh 1)	4
T6.2-70 (Sh 2)	4
T6.2-70 (Sh 3)	4
T6.2-70 (Sh 4)	4
T6.2-70 (Sh 5)	4
T6.2-71	4
T6.2-72	4
T6.2-73	4
T6.2-74	4
T6.2-75 (Sh 1)	4
T6.2-75 (Sh 2)	-
T6.2-76	-
T6.2-77	-
T6.2-78	-
T6.2-79 (Sh 1)	-
T6.2-79 (Sh 2)	-
T6.2-79 (Sh 3)	-
T6.2-79 (Sh 4)	-
T6.2-80	-
T6.2-81	-
T6.2-82	-
T6.2-83 (Sh 1)	5
T6.2-83 (Sh 2)	4
T6.2-83 (Sh 3)	4
T6.2-83 (Sh 4)	4
T6.2-83 (Sh 5)	4
T6.2-83 (Sh 6)	4
T6.2-83 (Sh 7)	4
T6.2-83 (Sh 8)	4
T6.2-83 (Sh 9)	4
T6.2-83 (Sh 10)	5
T6.2-83 (Sh 11)	4
T6.2-83 (Sh 12)	6
T6.2-83 (Sh 13)	5
T6.2-83 (Sh 14)	4
T6.2-84 (Sh 1)	-
T6.2-84 (Sh 2)	-
T6.2-85	-
T6.2-86	-
T6.2-87	7
T6.2-88	7
T6.2-89	4
T6.2-90	6
T6.2-91	-
T6.2-92 (Sh 1)	-
T6.2-92 (Sh 2)	-
T6.2-92 (Sh 3)	6
T6.2-92 (Sh 4)	2
T6.2-92 (Sh 5)	2
F6.2-1	-
F6.2-2	-
F6.2-3	-
F6.2-4	-
F6.2-5	-
F6.2-6	-
F6.2-7	-
F6.2-8	-
F6.2-9	-
F6.2-10	-
F6.2-11	-
F6.2-12	-
F6.2-13	-
F6.2-14	-
F6.2-15	-
F6.2-16	-
F6.2-17	-
F6.2-18	-
F6.2-19	-
F6.2-20	-
F6.2-21	-
F6.2-22	-
F6.2-23	-
F6.2-24	-
F6.2-25	-
F6.2-26	4
F6.2-27	4
F6.2-28	4
F6.2-29	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev. No.	Page No.	Rev. No.
F6.2-30	-	F6.2-94	-	Notes to F6.3-6 (Sh 11)	-
F6.2-31	-	F6.2-95	-	Notes to F6.3-6 (Sh 12)	-
F6.2-32	-	F6.2-96	4	Notes to F6.3-6 (Sh 13)	-
F6.2-33	-	F6.2-97	4	Notes to F6.3-6 (Sh 14)	-
F6.2-34	-			Notes to F6.3-6 (Sh 15)	-
F6.2-35	-	6.3-1	-	Notes to F6.3-6 (Sh 16)	-
F6.2-36	-	6.3-2	-	Notes to F6.3-6 (Sh 17)	-
F6.2-37	-	6.3-3	1	Notes to F6.3-6 (Sh 18)	-
F6.2-38	-	6.3-4	1	Notes to F6.3-6 (Sh 19)	-
F6.2-39	-	6.3-5	4	Notes to F6.3-6 (Sh 20)	-
F6.2-40	-	6.3-6	6	F6.3-7	-
F6.2-41	-	6.3-6a	6	F6.3-8	-
F6.2-42	-	6.3-7	1	F6.3-9	-
F6.2-43	4	6.3-8	1	F6.3-10	6
F6.2-44	4	6.3-9	4		
F6.2-45	4	6.3-9a	4	6.4-1	4
F6.2-46	4	6.3-10	1	6.4-2	-
F6.2-47	4	6.3-11	7	6.4-3	7
F6.2-48	4	6.3-12	1	6.4-4	2
F6.2-49	4	6.3-13	1	6.4-5	1
F6.2-50	4	6.3-14	1	6.4-6	7
F6.2-51	4	6.3-15	1	6.4-7	1
F6.2-52	4	6.3-16	1	6.4-8	7
F6.2-53	4	6.3-17	4	6.4-8a	5
F6.2-54 (Sh 1)	-	6.3-18	4	6.4-9	1
F6.2-54 (Sh 2)	-	6.3-19	1	6.4-10	7
F6.2-54 (Sh 3)	-	6.3-20	2	6.4-10a	7
F6.2-55	-	6.3-21	2	6.4-11	1
F6.2-56	-	6.3-22	1	6.4-12	6
F6.2-57 (Sh 1)	-	6.3-23	4	6.4-13	6
F6.2-57 (Sh 2)	-	6.3-24	1	6.4-14	1
F6.2-58 (Sh 1)	-	6.3-25	6	6.4-15	7
F6.2-58 (Sh 2)	-	6.3-26	1	6.4-16	7
F6.2-59	-	6.3-27	1	T6.4-1 (Sh 1)	7
F6.2-60	-	6.3-28	1	T6.4-1 (Sh 2)	7
F6.2-61	4	6.3-29	6	T6.4-1 (Sh 3)	7
F6.2-62	4	6.3-30	1	F6.4-1	1
F6.2-63	4	6.3-31	3	F6.4-2 (Sh 1)	1
F6.2-64	4	6.3-32	3	F6.4-2 (Sh 2)	1
F6.2-65	4	6.3-33	3		
F6.2-66	4	6.3-34	1	6.5-1	-
F6.2-67	4	6.3-35	1	6.5-2	-
F6.2-68	4	T6.3-1 (Sh 1)	4	6.5-3	-
F6.2-69 (Sh 1)	-	T6.3-1 (Sh 2)	2	6.5-4	-
F6.2-69 (Sh 2)	-	T6.3-2	7	6.5-5	-
F6.2-69 (Sh 3)	-	T6.3-3 (Sh 1)	-	6.5-6	-
F6.2-70 (Sh 1)	-	T6.3-3 (Sh 2)	1	6.5-7	-
F6.2-70 (Sh 2)	-	T6.3-4 (Sh 1)	-	6.5-8	-
F6.2-71	-	T6.3-4 (Sh 2)	-	6.5-9	-
F6.2-72	4	T6.3-5 (Sh 1)	-	6.5-10	-
F6.2-73	4	T6.3-5 (Sh 2)	-	6.5-11	-
F6.2-74	7	T6.3-5 (Sh 3)	-	6.5-12	-
F6.2-75	4	T6.3-5 (Sh 4)	-	6.5-13	7
F6.2-76	-	T6.3-5 (Sh 5)	-	6.5-14	-
F6.2-77	-	T6.3-5 (Sh 6)	-	6.5-15	7
F6.2-78	-	T6.3-5 (Sh 7)	-	6.5-16	4
F6.2-79	-	T6.3-5 (Sh 8)	-	6.5-17	7
F6.2-80	-	T6.3-5 (Sh 9)	-	6.5-18	4
F6.2-81	-	T6.3-6	-	6.5-19	-
F6.2-82	-	T6.3-7 (Sh 1)	-	6.5-20	-
F6.2-83	-	T6.3-7 (Sh 2)	4	T6.5-1 (Sh 1)	7
F6.2-84	-	T6.3-7 (Sh 2a)	4	T6.5-1 (Sh 2)	-
F6.2-85	-	T6.3-7 (Sh 3)	-	T6.5-1 (Sh 3)	7
F6.2-86	-	T6.3-8	1	T6.5-2 (Sh 1)	7
F6.2-87	-	T6.3-9	4	T6.5-2 (Sh 2)	-
F6.2-88	-	T6.3-10	-	T6.5-2 (Sh 3)	7
F6.2-89	-	F6.3-1	-	T6.5-3 (Sh 1)	7
F6.2-90	-	F6.3-2 (Sh 1)	7	T6.5-3 (Sh 2)	7
F6.2-91 (Sh 1)	5	F6.3-2 (Sh 2)	4	T6.5-3 (Sh 3)	7
F6.2-91 (Sh 2)	4	F6.3-3	5	T6.5-3 (Sh 4)	7
F6.2-91 (Sh 3)	4	F6.3-4	7	T6.5-4 (Sh 1)	7
F6.2-91 (Sh 4)	-	F6.3-5	6	T6.5-4 (Sh 2)	7
F6.2-91 (Sh 5)	-	F6.3-6 (Sh 1)	-	T6.5-4 (Sh 3)	7
F6.2-91 (Sh 6)	-	F6.3-6 (Sh 2)	-	T6.5-5 (Sh 1)	7
F6.2-91 (Sh 7)	5	Notes to F6.3-6 (Sh 3)	-	T6.5-5 (Sh 2)	7
F6.2-91 (Sh 8)	-	Notes to F6.3-6 (Sh 4)	-	T6.5-5 (Sh 3)	7
F6.2-91 (Sh 9)	5	Notes to F6.3-6 (Sh 5)	-	T6.5-6 (Sh 1)	7
F6.2-91 (Sh 10)	3	Notes to F6.3-6 (Sh 6)	-	T6.5-6 (Sh 2)	7
F6.2-91 (Sh 11)	2	Notes to F6.3-6 (Sh 7)	-	T6.5-6 (Sh 3)	7
F6.2-91 (Sh 12)	2	Notes to F6.3-6 (Sh 8)	-	T6.5-7	-
F6.2-92	4	Notes to F6.3-6 (Sh 9)	-	T6.5-8	-
F6.2-93	-	Notes to F6.3-6 (Sh 10)	-	F6.5-1	5

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
F6.5-2	-
F6.5-3	-
6.6-1	7
6.6-2	6
6.6-3	7
6.6-4	7
6.7-1	-
6.8-1	6
6.8-1a	3
6.8-2	3
6.8-2a	3
6.8-3	4
6.8-3a	4
6.8-4	6
6.8-4a	5
6.8-5	1
6.8-6	7
6.8-7	5
T6.8-1	-
T6.8-2	-
F6.8-1	4
F6.8-2	7
F6.8-3	-
Appendix 6A	
Cover	6
All Other Sheets	-
VOLUME 8	
Chapter 7	
7.0-1	5
7.0-2	6
7.0-3	1
7.0-4	5
7.0-5	5
7.0-6	4
7.0-7	5
7.0-8	5
7.1-1	7
7.1-2	-
7.1-3	-
7.1-4	-
7.1-5	-
7.1-6	6
7.1-7	7
7.1-8	6
7.1-8a	4
7.1-9	-
7.1-10	6
7.1-10a	6
7.1-11	4
7.1-12	7
7.1-13	-
7.1-14	-
7.1-15	7
7.1-16	7
7.1-17	6
7.1-18	-
7.1-19	7
7.1-20	6
7.1-20a	6
7.1-21	7
7.1-22	7
7.1-23	7
7.1-23a	3
7.1-24	-
7.1-25	-
7.1-26	-
7.1-27	7
7.1-27a	7
7.1-28	4
7.1-28a	5
7.1-29	4
7.1-29a	4

Page No.	Rev. No.
7.1-30	7
7.1-30a	5
7.1-31	5
7.1-32	7
T7.1-1 (Sh 1)	-
T7.1-1 (Sh 2)	-
T7.1-1 (Sh 3)	-
T7.1-1 (Sh 4)	-
T7.1-1 (Sh 5)	-
T7.1-1 (Sh 6)	5
T7.1-1 (Sh 7)	-
T7.1-1 (Sh 8)	-
T7.1-1 (Sh 9)	5
T7.1-1 (Sh 10)	-
T7.1-1 (Sh 11)	-
T7.1-1 (Sh 12)	-
F7.1-1	4
F7.1-2	-
7.2-1	4
7.2-2	6
7.2-2a	6
7.2-3	4
7.2-4	7
7.2-5	4
7.2-6	4
7.2-7	7
7.2-8	4
7.2-9	7
7.2-10	7
7.2-11	4
7.2-12	4
7.2-13	7
7.2-13a	7
7.2-14	4
7.2-15	4
7.2-16	7
7.2-17	4
7.2-18	4
7.2-19	7
7.2-20	6
7.2-21	6
7.2-22	7
7.2-23	4
7.2-24	7
7.2-24a	5
7.2-25	7
7.2-26	7
7.2-27	4
7.2-28	4
7.2-29	7
7.2-30	7
7.2-31	7
7.2-32	4
7.2-33	5
7.2-33a	5
7.2-34	4
7.2-35	4
7.2-36	4
7.2-37	5
7.2-38	4
7.2-39	7
7.2-40	5
T7.2-1 (Sh 1)	-
T7.2-1 (Sh 2)	-
T7.2-2 (Sh 1)	-
T7.2-2 (Sh 2)	-
T7.2-3 (Sh 1)	7
T7.2-3 (Sh 2)	7
T7.2-4 (Sh 1)	7
T7.2-4 (Sh 2)	4
T7.2-4 (Sh 3)	7
T7.2-4 (Sh 3a)	4
T7.2-4 (Sh 4)	-
T7.2-4 (Sh 5)	-
T7.2-4 (Sh 6)	-
F7.2-1	4
F7.2-2	4
F7.2-3	4
F7.2-4	4
F7.2-5 (Sh 1)	4
F7.2-5 (Sh 2)	4

Page No.	Rev. No.
F7.2-6	7
F7.2-7 (Sh 1)	4
F7.2-7 (Sh 2)	4
F7.2-8	7
F7.2-9	2
F7.2-10	-
F7.2-11	1
F7.2-12	-
F7.2-13 (Sh 1)	7
F7.2-13 (Sh 2)	4
F7.2-14	6
F7.2-15	4
F7.2-16	4
7.3-1	7
7.3-2	7
7.3-3	4
7.3-4	7
7.3-5	7
7.3-6	7
7.3-7	4
7.3-8	4
7.3-9	4
7.3-10	6
7.3-11	7
7.3-12	7
7.3-13	7
7.3-14	4
7.3-15	4
7.3-16	7
7.3-17	7
7.3-18	4
7.3-19	7
7.3-20	7
7.3-21	7
7.3-22	7
7.3-23	7
7.3-23a	7
7.3-24	4
7.3-25	6
7.3-26	6
7.3-27	6
7.3-27a	6
7.3-28	6
7.3-29	4
T7.3-1 (Sh 1)	-
T7.3-1 (Sh 2)	-
T7.3-1 (Sh 3)	-
T7.3-2 (Sh 1)	6
T7.3-2 (Sh 2)	1
F7.3-1	4
F7.3-2	7
7.4-1	-
7.4-2	1
7.4-3	-
7.4-4	3
7.4-4a	3
7.4-5	-
7.4-6	-
7.4-7	-
7.4-8	-
T7.4-1 (Sh 1)	1
T7.4-1 (Sh 2)	3
T7.4-1 (Sh 2a)	3
T7.4-1 (Sh 3)	1
T7.4-1 (Sh 4)	3
T7.4-1 (Sh 5)	3
T7.4-1 (Sh 6)	3
T7.4-1 (Sh 7)	3
T7.4-1 (Sh 8)	3
T7.4-1 (Sh 9)	7
7.5-1	-
7.5-2	6
7.5-2a	6
7.5-3	7
7.5-4	-
7.5-5	-
7.5-6	-
7.5-7	7
7.5-8	7

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
7.5-9	7
7.5-10	-
7.5-11	-
7.5-12	-
T7.5-1 (Sh 1)	7
T7.5-1 (Sh 2)	1
T7.5-1 (Sh 3)	7
T7.5-1 (Sh 4)	7
T7.5-1 (Sh 5)	7
T7.5-1 (Sh 6)	7
T7.5-1 (Sh 7)	7
T7.5-1 (Sh 8)	1
T7.5-1 (Sh 9)	7
T7.5-1 (Sh 10)	7
T7.5-1 (Sh 11)	1
T7.5-1 (Sh 12)	1
T7.5-1 (Sh 13)	1
T7.5-1 (Sh 14)	-
T7.5-1 (Sh 15)	1
T7.5-1 (Sh 16)	-
T7.5-1 (Sh 17)	3
T7.5-1 (Sh 18)	3
T7.5-1 (Sh 19)	7
T7.5-1 (Sh 20)	1
T7.5-1 (Sh 21)	7
T7.5-1 (Sh 22)	-
T7.5-1 (Sh 23)	7
T7.5-1 (Sh 23a)	7
T7.5-1 (Sh 24)	7
T7.5-1 (Sh 25)	6
T7.5-1 (Sh 26)	-
T7.5-1 (Sh 27)	5
T7.5-1 (Sh 28)	1
T7.5-1 (Sh 29)	7
T7.5-1 (Sh 30)	-
T7.5-1 (Sh 31)	7
T7.5-1 (Sh 32)	-
T7.5-1 (Sh 33)	7
T7.5-1 (Sh 34)	-
T7.5-1 (Sh 35)	7
T7.5-1 (Sh 36)	-
T7.5-1 (Sh 37)	7
T7.5-1 (Sh 37a)	7
T7.5-2 (Sh 1)	6
T7.5-2 (Sh 2)	6
T7.5-2 (Sh 3)	7
T7.5-2 (Sh 4)	-
T7.5-2 (Sh 5)	7
T7.5-2 (Sh 6)	2
7.6-1	7
7.6-2	7
7.6-3	7
7.6-4	1
7.6-5	7
7.6-6	7
7.6-7	7
7.6-8	7
7.6-9	7
7.6-10	6
7.6-11	7
F7.6-1	1
F7.6-2	1
F7.6-3	7
F7.6-4 (Sh 1)	7
F7.6-4 (Sh 2)	7
F7.6-5	-
F7.6-6	-
F7.6-7	-
7.7-1	-
7.7-2	7
7.7-3	2
7.7-4	-
7.7-5	-
7.7-6	-
7.7-7	-
7.7-8	-
7.7-9	4
7.7-10	7
7.7-11	3
7.7-12	-

Page No.	Rev. No.
7.7-13	7
7.7-14	7
7.7-15	7
7.7-16	3
7.7-17	2
7.7-18	7
7.7-19	7
7.7-20	7
7.7-20a	7
7.7-21	7
7.7-22	5
7.7-23	7
7.7-24	5
7.7-25	5
7.7-26	5
7.7-27	5
7.7-28	7
T7.7-1 (Sh 1)	2
T7.7-1 (Sh 2)	4
T7.7-2	5
F7.7-1	2
F7.7-2	-
F7.7-3	-
F7.7-4	-
F7.7-5	-
F7.7-6	-
F7.7-7	4
F7.7-8	7
F7.7-9	-
F7.7-10	5
F7.7-11	5
F7.7-12	5
F7.7-13	5
F7.7-14	7
F7.7-15	-

Appendix 7A

Page No.	Rev. No.
7A-1	-
7A-2	-
7A-3	-
7A-4	-
7A-5	-
7A-6	-
7A-7	-
7A-8	-
7A-9	-
7A-10	-
7A-11	-
7A-12	-
7A-13	-
7A-14	1
7A-15	-
7A-16	-
7A-17	-
7A-18	1
7A-19	-
7A-20	-
7A-21	-
7A-22	-
7A-23	-
7A-24	-
7A-25	6
7A-26	-
7A-27	-
7A-28	-
7A-29	-
7A-30	-
7A-31	3
7A-32	-
7A-33	7

Chapter 8

Page No.	Rev. No.
8.0-1	2
8.0-2	3
8.0-2a	3
8.0-3	-
8.0-4	-
8.0-5	-
8.0-6	-
8.0-7	-
8.0-8	6
8.1-1	-
8.1-2	-
8.1-3	7
8.1-4	7
8.1-5	7
8.1-6	7
8.1-7	7
8.1-8	7
8.1-9	7
8.1-10	7
8.1-11	7
8.1-12	7
8.1-13	7
8.1-14	7
8.1-15	7
8.1-16	7
8.1-17	7
T8.1-1 (Sh 1)	-
T8.1-1 (Sh 2)	-
T8.1-1 (Sh 3)	-
F8.1-1 (Sh 1)	-
F8.1-1 (Sh 2)	6
8.2-1	-
8.2-2	-
8.2-3	6
8.2-3a	6
8.2-4	-
8.2-5	6
8.2-5a	6
8.2-6	-
8.2-7	-
8.2-8	-
8.2-9	6
8.2-9a	6
8.2-10	-
8.2-11	7
8.2-11a	7
8.2-12	-
8.2-13	-
8.2-14	-
8.2-15	6
8.2-15a	6
8.2-16	7
8.2-17	2
8.2-18	2
8.2-19	2
8.2-20	3
8.2-20a	3
8.2-21	2
8.2-22	2
T8.2-1 (Sh 1)	-
T8.2-1 (Sh 2)	-
T8.2-1 (Sh 3)	-
T8.2-1 (Sh 4)	-
F8.2-1	-
F8.2-2 (Sh 1)	-
F8.2-2 (Sh 2)	-
F8.2-3	-
F8.2-4	-
F8.2-5	6
F8.2-6	-
F8.2-7	-
F8.2-8	2
F8.2-9 (Sh 1)	-
F8.2-9 (Sh 2)	-
F8.2-10	2
F8.2-11	7
8.3-1	-
8.3-2	-
8.3-3	7
8.3-3a	7
8.3-4	-
8.3-5	2
8.3-6	2
8.3-6a	2
8.3-7	7
8.3-8	7
8.3-8a	7

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
8.3-8b	7
8.3-9	-
8.3-10	-
8.3-11	7
8.3-12	7
8.3-12a	3
8.3-13	4
8.3-14	-
8.3-15	-
8.3-16	-
8.3-17	-
8.3-18	-
8.3-19	-
8.3-20	3
8.3-21	3
8.3-22	-
8.3-23	6
8.3-24	-
8.3-25	-
8.3-26	7
8.3-26a	7
8.3-27	-
8.3-28	-
8.3-29	-
8.3-30	3
8.3-31	7
8.3-32	6
8.3-32a	6
8.3-33	-
8.3-34	-
8.3-35	-
8.3-36	-
8.3-37	-
8.3-38	7
8.3-39	7
8.3-40	6
8.3-40a	6
8.3-41	3
8.3-42	6
8.3-42a	6
8.3-43	3
8.3-44	3
8.3-45	3
8.3-46	3
8.3-47	3
8.3-48	3
8.3-48a	3
8.3-49	-
8.3-50	-
8.3-51	6
8.3-52	-
8.3-53	4
8.3-54	-
8.3-55	7
8.3-55a	7
8.3-56	6
8.3-56a	6
8.3-57	-
8.3-58	6
8.3-59	7
8.3-60	6
8.3-61	6
8.3-61a	7
8.3-62	4
8.3-63	4
8.3-63a	3
8.3-64	-
8.3-65	7
8.3-65a	6
8.3-66	7
8.3-67	6
T8.3-1 (Sh 1)	7
T8.3-1 (Sh 2)	7
T8.3-1 (Sh 3)	7
T8.3-1 (Sh 4)	7
T8.3-1 (Sh 5)	7
T8.3-2 (Sh 1)	7
T8.3-2 (Sh 2)	7
T8.3-2 (Sh 3)	7
T8.3-2 (Sh 4)	7
T8.3-2 (Sh 5)	7
T8.3-2 (Sh 6)	7

Page No.	Rev. No.
T8.3-3 (Sh 1)	-
T8.3-3 (Sh 2)	-
T8.3-3 (Sh 3)	-
T8.3-3 (Sh 4)	-
T8.3-3 (Sh 5)	-
T8.3-3 (Sh 6)	-
T8.3-4	-
T8.3-5	6
T8.3-6	4
T8.3-7 (Sh 1)	-
T8.3-7 (Sh 2)	4
T8.3-7 (Sh 3)	4
T8.3-7 (Sh 4)	-
T8.3-7 (Sh 5)	-
T8.3-7 (Sh 6)	-
T8.3-7 (Sh 7)	-
T8.3-7 (Sh 8)	-
T8.3-8 (Sh 1)	6
T8.3-8 (Sh 2)	6
T8.3-9	3
T8.3-10 (Sh 1)	6
T8.3-10 (Sh 2)	6
F8.3-1	7
F8.3-2	4
F8.3-3	6
F8.3-4 (Sh 1)	4
F8.3-4 (Sh 2)	-
F8.3-5	7
F8.3-6	7
F8.3-7 (Sh 1)	-
F8.3-7 (Sh 2)	-
F8.3-8	4
F8.3-9	4
F8.3-10 (Sh 1)	4
F8.3-10 (Sh 2)	4
F8.3-11	7
F8.3-12	5
F8.3-13	7
F8.3-14	6
F8.3-15	-
F8.3-16	5
F8.3-17 (Sh 1)	7
F8.3-17 (Sh 2)	5
F8.3-18	7
F8.3-19	7
F8.3-20	7
F8.3-21	-
F8.3-22	5
F8.3-23 (Sh 1)	7
F8.3-23 (Sh 2)	6
F8.3-24	-
F8.3-25	7
F8.3-26	-
F8.3-27	5
F8.3-28	7
F8.3-29	7
F8.3-30	7
F8.3-31	7
F8.3-32	7
F8.3-33	7
F8.3-34	7
F8.3-35	7
F8.3-36	5
F8.3-37 (Sh 1)	6
F8.3-37 (Sh 2)	5
F8.3-38 (Sh 1)	6
F8.3-38 (Sh 2)	6
F8.3-39	6
F8.3-40	6
F8.3-41	6
F8.3-42	6
F8.3-43	5
F8.3-44	5
F8.3-45	4
F8.3-46	7
F8.3-47	-
F8.3-48	-
F8.3-49	-
F8.3-50	-
F8.3-51 (Sh 1)	6
F8.3-51 (Sh 2)	-
F8.3-52 (Sh 1)	6

Page No.	Rev. No.
F8.3-52 (Sh 2)	-
F8.3-53 (Sh 1)	6
F8.3-53 (Sh 2)	-
F8.3-54	3
F8.3-55	-
F8.3-56	1
F8.3-57	-
F8.3-58	6
F8.3-59	6
F8.3-60	7
F8.3-61	7
8.4-1	6
8.4-2	3
8.4-3	3
Appendix 8A	
All Sheets	-
VOLUME 9	
Chapter 9	
9.0-1	6
9.0-2	6
9.0-3	6
9.0-4	5
9.0-5	5
9.0-6	5
9.0-7	5
9.0-8	6
9.0-9	6
9.0-10	7
9.0-11	7
9.0-12	7
9.0-13	1
9.0-14	6
9.0-15	6
9.0-16	1
9.0-17	4
9.0-18	7
9.0-19	7
9.1-1	7
9.1-2	7
9.1-3	1
9.1-4	1
9.1-5	1
9.1-6	7
9.1-6a	7
9.1-7	6
9.1-8	6
9.1-9	6
9.1-10	6
9.1-11	6
9.1-12	6
9.1-13	7
9.1-14	7
9.1-15	7
9.1-16	7
9.1-17	7
9.1-17a	7
9.1-18	6
9.1-19	6
9.1-20	7
9.1-20a	7
9.1-21	7
9.1-22	6
9.1-23	6
9.1-24	7
9.1-24a	7
9.1-25	6
9.1-26	6
9.1-27	7
9.1-28	7
9.1-28a	7
9.1-29	6
9.1-30	7
9.1-31	7
9.1-32	6
9.1-33	6
9.1-34	6
9.1-35	6

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.	Page No.	Rev. No.	Page No.	Rev. No.
9.1-36	6	9.2-27a	4	9.3-5	-
9.1-37	6	9.2-28	1	9.3-6	3
9.1-38	7	9.2-29	1	9.3-7	4
9.1-39	7	9.2-30	6	9.3-7a	4
9.1-39a	7	9.2-31	6	9.3-8	4
9.1-40	6	9.2-32	1	9.3-9	-
9.1-41	6	9.2-33	6	9.3-10	3
9.1-42	6	9.2-33a	6	9.3-11	-
9.1-43	6	9.2-34	7	9.3-12	7
9.1-44	6	9.2-35	6	9.3-12a	2
9.1-45	7	9.2-36	6	9.3-13	7
9.1-45a	7	9.2-37	7	9.3-14	3
9.1-46	7	9.2-37a	3	9.3-15	7
9.1-46a	7	9.2-38	1	9.3-15a	2
9.1-47	6	9.2-39	1	9.3-16	-
9.1-48	6	9.2-40	6	9.3-17	-
T9.1-1 (Sh 1)	7	9.2-41	6	9.3-18	3
T9.1-1 (Sh 2)	7	T9.2-1	5	9.3-19	-
T9.1-2 (Sh 1)	6	T9.2-2	6	9.3-20	-
T9.1-2 (Sh 2)	1	T9.2-3 (Sh 1)	-	9.3-21	-
T9.1-3	7	T9.2-3 (Sh 2)	-	9.3-22	-
T9.1-4	6	T9.2-3 (Sh 3)	-	9.3-23	7
T9.1-5	6	T9.2-3 (Sh 4)	-	9.3-24	7
T9.1-6	6	T9.2-3 (Sh 5)	-	9.3-25	-
F9.1-1	5	T9.2-4	6	9.3-26	-
F9.1-2 (Sh 1)	7	T9.2-5 (Sh 1)	6	9.3-27	-
F9.1-2 (Sh 2)	5	T9.2-5 (Sh 2)	6	9.3-28	-
F9.1-2 (Sh 3)	7	T9.2-5 (Sh 3)	6	9.3-29	-
F9.1-3	6	T9.2-6 (Sh 1)	6	9.3-30	-
F9.1-4	1	T9.2-6 (Sh 2)	-	9.3-31	3
F9.1-5	1	T9.2-6 (Sh 3)	6	9.3-31a	3
F9.1-6 (Sh 1)	6	T9.2-7 (Sh 1)	5	9.3-32	7
F9.1-6 (Sh 2)	6	T9.2-7 (Sh 2)	-	9.3-33	6
F9.1-6 (Sh 3)	6	T9.2-7 (Sh 3)	-	9.3-34	6
F9.1-7	-	T9.2-7 (Sh 4)	-	9.3-35	-
F9.1-8	-	T9.2-8	-	9.3-36	5
F9.1-9	-	T9.2-9 (Sh 1)	-	9.3-37	-
F9.1-10	-	T9.2-9 (Sh 2)	-	9.3-38	-
F9.1-11	-	T9.2-9 (Sh 3)	-	9.3-39	-
F9.1-12	6	T9.2-10	5	9.3-40	-
F9.1-13	1	T9.2-11 (Sh 1)	5	9.3-41	-
F9.1-14	1	T9.2-11 (Sh 2)	5	9.3-42	5
F9.1-15	1	T9.2-11 (Sh 3)	6	9.3-43	-
F9.1-16	1	T9.2-12 (Sh 1)	-	9.3-44	3
F9.1-17	1	T9.2-12 (Sh 2)	-	9.3-45	-
F9.1-18	1	T9.2-13	1	9.3-46	-
F9.1-19	6	T9.2-14	-	9.3-47	6
F9.1-20	6	T9.2-15 (Sh 1)	-	9.3-47a	6
F9.1-21	6	T9.2-15 (Sh 2)	-	9.3-48	4
F9.1-22	6	T9.2-16	7	9.3-48a	4
		T9.2-17	6	9.3-49	-
9.2-1	6	F9.2-1	6	9.3-50	-
9.2-2	1	F9.2-2 (Sh 1)	7	9.3-51	-
9.2-3	1	F9.2-2 (Sh 2)	7	9.3-52	-
9.2-4	4	F9.2-2 (Sh 3)	4	9.3-53	-
9.2-5	6	F9.2-3	5	9.3-54	-
9.2-5a	6	F9.2-4 (Sh 1)	6	9.3-55	7
9.2-6	1	F9.2-4 (Sh 2)	5	9.3-55a	7
9.2-7	1	F9.2-4 (Sh 3)	6	9.3-56	-
9.2-8	1	F9.2-4 (Sh 4)	2	9.3-57	7
9.2-9	7	F9.2-5	5	9.3-58	7
9.2-9a	7	F9.2-6 (Sh 1)	6	9.3-59	7
9.2-10	6	F9.2-6 (Sh 2)	2	9.3-60	3
9.2-10a	6	F9.2-6 (Sh 3)	7	9.3-60a	6
9.2-11	1	F9.2-6 (Sh 4)	5	9.3-61	5
9.2-12	1	F9.2-6 (Sh 5)	-	9.3-61a	5
9.2-13	1	F9.2-7	4	9.3-62	-
9.2-14	1	F9.2-8	7	9.3-63	3
9.2-15	1	F9.2-9 (Sh 1)	7	9.3-63a	3
9.2-16	1	F9.2-9 (Sh 2)	7	9.3-64	-
9.2-17	1	F9.2-10	-	9.3-65	-
9.2-18	2	F9.2-11	7	9.3-66	-
9.2-19	1	F9.2-12	5	9.3-67	5
9.2-20	5	F9.2-13	3	9.3-67a	5
9.2-21	5	F9.2-14	6	9.3-68	-
9.2-22	6			9.3-69	-
9.2-23	6	9.3-1	6	9.3-70	2
9.2-23a	6	9.3-2	6	9.3-70a	2
9.2-24	1	9.3-2a	6	9.3-71	-
9.2-25	1	9.3-3	6	9.3-72	3
9.2-26	6	9.3-4	3	9.3-73	-
9.2-27	4	9.3-4a	3	9.3-74	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
9.3-75	6
9.3-76	7
9.3-77	7
T9.3-1 (Sh 1)	4
T9.3-1 (Sh 2)	4
T9.3-1 (Sh 3)	4
T9.3-1 (Sh 4)	7
T9.3-1 (Sh 5)	7
T9.3-1 (Sh 6)	7
T9.3-1 (Sh 7)	7
T9.3-1 (Sh 8)	4
T9.3-2 (Sh 1)	7
T9.3-2 (Sh 2)	7
T9.3-2 (Sh 3)	7
T9.3-2 (Sh 4)	7
T9.3-3 (Sh 1)	-
T9.3-3 (Sh 2)	-
T9.3-4	-
T9.3-5	-
T9.3-6 (Sh 1)	-
T9.3-6 (Sh 2)	-
T9.3-6 (Sh 3)	-
T9.3-6 (Sh 4)	-
T9.3-6 (Sh 5)	-
T9.3-6 (Sh 6)	6
T9.3-6 (Sh 7)	-
T9.3-6 (Sh 8)	-
T9.3-6 (Sh 9)	-
T9.3-6 (Sh 10)	-
T9.3-6 (Sh 11)	-
T9.3-7 (Sh 1)	4
T9.3-7 (Sh 2)	4
T9.3-7 (Sh 3)	4
T9.3-7 (Sh 4)	4
T9.3-7 (Sh 5)	4
T9.3-7 (Sh 6)	4
T9.3-7 (Sh 7)	4
T9.3-7 (Sh 8)	4
T9.3-7 (Sh 9)	4
T9.3-7 (Sh 10)	4
T9.3-7 (Sh 11)	4
T9.3-7 (Sh 12)	4
T9.3-7 (Sh 13)	4
T9.3-7 (Sh 14)	4
T9.3-7 (Sh 15)	4
T9.3-7 (Sh 16)	4
T9.3-7 (Sh 17)	4
T9.3-7 (Sh 18)	4
T9.3-7 (Sh 19)	4
T9.3-7 (Sh 20)	4
T9.3-8 (Sh 1)	-
T9.3-8 (Sh 2)	-
T9.3-8 (Sh 3)	-
T9.3-8 (Sh 4)	-
T9.3-8 (Sh 5)	-
T9.3-8 (Sh 6)	-
T9.3-8 (Sh 7)	-
T9.3-9 (Sh 1)	-
T9.3-9 (Sh 2)	-
T9.3-10	-
T9.3-11 (Sh 1)	-
T9.3-11 (Sh 2)	-
F9.3-1	2
F9.3-2 (Sh 1)	7
F9.3-2 (Sh 2)	5
F9.3-2 (Sh 3)	7
F9.3-3	7
F9.3-4	4
F9.3-5	5
F9.3-6 (Sh 1)	7
F9.3-6 (Sh 2)	4
F9.3-6 (Sh 3)	7
F9.3-6 (Sh 4)	6
F9.3-7	5
F9.3-8	7
F9.3-9 (Sh 1)	6
F9.3-9 (Sh 2)	4
F9.3-9 (Sh 3)	4
F9.3-10	-
F9.3-11	5
F9.3-12	7
F9.3-13	4
F9.3-14	6
F9.3-15	4
F9.3-16	-
F9.3-17	5
F9.3-18	5
F9.3-19	5
F9.3-20	3
F9.3-21 (Sh 1)	7
F9.3-21 (Sh 2)	3
F9.3-22	-
F9.3-23 (Sh 1)	4
F9.3-23 (Sh 2)	7
F9.3-23 (Sh 3)	3
F9.3-24	-
F9.3-25	4
F9.3-26	7
F9.3-27	7
F9.3-28	7
F9.3-29	6
F9.3-30	7
F9.3-31 (Sh 1)	7
F9.3-31 (Sh 2)	4
F9.3-32	6
F9.3-33	7
F9.3-34	-
F9.3-35	2
F9.3-36	2
F9.3-37 (Sh 1)	4
F9.3-37 (Sh 2)	6
F9.3-37 (Sh 3)	-
F9.3-38	6
F9.3-39 (Sh 1)	7
F9.3-39 (Sh 2)	7
F9.3-40	6
F9.3-41	7
F9.3-42	6
9.4-1	3
9.4-2	7
9.4-3	7
9.4-4	7
9.4-5	7
9.4-6	7
9.4-7	7
9.4-8	7
9.4-9	7
9.4-10	7
9.4-10a	7
9.4-10b	7
9.4-10c	7
9.4-10d	7
9.4-11	4
9.4-12	4
9.4-13	2
9.4-14	2
9.4-15	2
9.4-16	2
9.4-17	2
9.4-18	2
9.4-19	2
9.4-20	2
9.4-21	2
9.4-22	2
9.4-23	2
9.4-24	1
9.4-25	1
9.4-26	6
9.4-27	3
9.4-28	1
9.4-29	1
9.4-30	6
9.4-31	7
9.4-31a	6
9.4-32	1
9.4-33	1
9.4-34	2
9.4-35	1
9.4-36	1
9.4-37	1
9.4-38	2
9.4-39	1
9.4-40	1
9.4-41	1
9.4-42	3
9.4-43	6
9.4-44	1
9.4-45	1
9.4-46	1
9.4-47	2
9.4-48	4
9.4-49	2
9.4-50	3
9.4-51	1
9.4-52	1
9.4-53	1
9.4-54	1
9.4-55	1
9.4-56	1
9.4-57	2
9.4-58	2
9.4-59	2
9.4-60	2
9.4-61	7
9.4-62	3
9.4-63	2
9.4-64	2
T9.4-1	7
T9.4-1A (Sh 1)	7
T9.4-1A (Sh 2)	7
T9.4-2 (Sh 1)	-
T9.4-2 (Sh 2)	-
T9.4-2 (Sh 3)	-
T9.4-3 (Sh 1)	-
T9.4-3 (Sh 2)	-
T9.4-4	-
T9.4-5 (Sh 1)	-
T9.4-5 (Sh 2)	-
T9.4-5 (Sh 3)	6
T9.4-5 (Sh 4)	-
T9.4-6 (Sh 1)	-
T9.4-6 (Sh 2)	-
T9.4-7	-
T9.4-8 (Sh 1)	7
T9.4-8 (Sh 2)	7
T9.4-9 (Sh 1)	-
T9.4-9 (Sh 2)	-
T9.4-9 (Sh 3)	-
T9.4-9 (Sh 4)	-
T9.4-9 (Sh 5)	-
T9.4-10	-
T9.4-11	-
T9.4-12 (Sh 1)	-
T9.4-12 (Sh 2)	4
T9.4-12 (Sh 3)	-
T9.4-13 (Sh 1)	-
T9.4-13 (Sh 2)	-
T9.4-14 (Sh 1)	-
T9.4-14 (Sh 2)	-
T9.4-15	-
T9.4-16 (Sh 1)	-
T9.4-16 (Sh 2)	-
T9.4-16 (Sh 3)	-
T9.4-17 (Sh 1)	7
T9.4-17 (Sh 2)	7
T9.4-17 (Sh 3)	7
T9.4-17 (Sh 4)	7
T9.4-17a	7
T9.4-18 (Sh 1)	-
T9.4-18 (Sh 2)	-
T9.4-19 (Sh 1)	3
T9.4-19 (Sh 1a)	3
T9.4-19 (Sh 2)	-
T9.4-19 (Sh 3)	7
T9.4-20 (Sh 1)	6
T9.4-20 (Sh 2)	6
T9.4-20 (Sh 3)	6
T9.4-20 (Sh 4)	6
T9.4-21 (Sh 1)	-
T9.4-21 (Sh 2)	-
T9.4-21 (Sh 3)	-
T9.4-22 (Sh 1)	6
T9.4-22 (Sh 2)	-
T9.4-22 (Sh 3)	6
T9.4-23 (Sh 1)	6

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
T9.4-23 (Sh 2)	-
T9.4-23 (Sh 3)	6
F9.4-1	7
F9.4-2	7
F9.4-3	7
F9.4-4	7
F9.4-5	4
F9.4-6	6
F9.4-7	6
F9.4-8	6
F9.4-9 (Sh 1)	4
F9.4-9 (Sh 2)	7
F9.4-10	-
F9.4-11 (Sh 1)	-
F9.4-11 (Sh 2)	6
F9.4-11 (Sh 3)	-
F9.4-11 (Sh 4)	-
F9.4-12	2
F9.4-13 (Sh 1)	7
F9.4-13 (Sh 2)	5
F9.4-14	7
F9.4-15	7
F9.4-16	7
F9.4-17 (Sh 1)	-
F9.4-17 (Sh 2)	-
F9.4-18	5
F9.4-19	7
F9.4-20	6
F9.4-21	2
F9.4-22	6
F9.4-23	4
F9.4-24 (Sh 1)	7
F9.4-24 (Sh 2)	4
F9.4-24 (Sh 3)	6
F9.4-25	7
F9.4-26	7
F9.4-27	7
9.5-1	-
9.5-2	6
9.5-3	7
9.5-4	6
9.5-4a	6
9.5-5	-
9.5-6	6
9.5-6a	6
9.5-7	-
9.5-8	7
9.5-8a	7
9.5-9	-
9.5-10	3
9.5-11	-
9.5-12	7
9.5-13	7
9.5-13a	6
9.5-14	-
9.5-15	5
9.5-15a	6
9.5-16	6
9.5-16a	6
9.5-17	6
9.5-18	6
9.5-19	5
9.5-20	7
9.5-20a	7
9.5-21	3
9.5-22	-
9.5-23	-
9.5-24	-
9.5-25	2
9.5-26	-
9.5-27	7
9.5-27a	7
9.5-28	6
9.5-29	-
9.5-30	-
9.5-31	7
9.5-32	6
9.5-33	6
9.5-34	-
9.5-35	6
9.5-36	6

Rev. Page No.	No.
9.5-37	6
9.5-38	6
9.5-39	6
9.5-40	6
9.5-41	6
9.5-41a	6
9.5-42	-
9.5-43	-
9.5-44	-
9.5-45	7
9.5-46	1
9.5-47	2
9.5-48	2
9.5-49	2
9.5-50	6
9.5-51	2
T9.5-1 (Sh 1)	-
T9.5-1 (Sh 2)	3
T9.5-1 (Sh 2a)	3
T9.5-2 (Sh 1)	3
T9.5-2 (Sh 2)	6
T9.5-3 (Sh 1)	1
T9.5-3 (Sh 2)	1
T9.5-4	-
T9.5-5	-
T9.5-6 (Sh 1)	-
T9.5-6 (Sh 2)	-
T9.5-7	-
T9.5-8	-
T9.5-9	4
F9.5-1	7
F9.5-2	7
F9.5-3	6
F9.5-4	7
F9.5-5	7
F9.5-6	7
F9.5-7 (Sh 1)	6
F9.5-7 (Sh 2)	6
F9.5-8	7
F9.5-9 (Sh 1)	-
F9.5-9 (Sh 2)	-
F9.5-10 (Sh 1)	7
F9.5-10 (Sh 2)	7
F9.5-11 (Sh 1)	7
F9.5-11 (Sh 2)	7
F9.5-12 (Sh 1)	7
F9.5-12 (Sh 2)	7
F9.5-13 (Sh 1)	7
F9.5-13 (Sh 2)	7
F9.5-14 (Sh 1)	7
F9.5-14 (Sh 2)	7

VOLUME 10

Chapter 10

Page No.	No.
10.0-1	-
10.0-2	3
10.0-3	6
10.0-4	1
10.0-5	7
10.0-6	-
10.0-7	-
10.1-1	7
10.1-2	-
F10.1-1	-
F10.1-2	-
10.2-1	7
10.2-2	-
10.2-3	1
10.2-4	5
10.2-5	-
10.2-6	6
10.2-6a	6
10.2-7	-
10.2-8	6
10.2-9	-
10.2-10	-
10.2-11	5
F10.2-1	5

Rev. Page No.	No.
10.3-1	-
10.3-2	-
10.3-3	6
10.3-4	6
10.3-5	3
10.3-6	-
10.3-7	7
10.3-7a	4
10.3-8	7
10.3-9	7
10.3-10	-
10.3-11	-
T10.3-1	7
F10.3-1	6
F10.3-2 (Sh 1)	6
F10.3-2 (Sh 2)	6
F10.3-3	7
F10.3-4	6
F10.3-5	6
F10.3-6	4
F10.3-7	4
F10.3-8	7
F10.3-9	2
F10.3-10	-
F10.3-11	5
10.4-1	-
10.4-2	6
10.4-3	7
10.4-4	-
10.4-5	7
10.4-6	6
10.4-7	3
10.4-7a	3
10.4-8	-
10.4-9	-
10.4-10	-
10.4-11	-
10.4-12	6
10.4-12a	6
10.4-13	-
10.4-14	5
10.4-15	-
10.4-16	7
10.4-17	3
10.4-18	-
10.4-19	4
10.4-19a	4
10.4-20	7
10.4-21	-
10.4-22	6
10.4-23	-
10.4-24	7
10.4-25	3
10.4-26	7
10.4-27	7
10.4-28	7
10.4-29	7
10.4-30	1
10.4-31	7
10.4-32	7
10.4-33	7
10.4-34	1
10.4-35	7
10.4-35a	7
10.4-36	1
10.4-37	5
10.4-38	1
10.4-39	3
10.4-40	3
10.4-41	1
10.4-42	4
10.4-42a	4
10.4-43	1
10.4-44	6
10.4-45	1
T10.4-1 (Sh 1)	-
T10.4-1 (Sh 2)	6
T10.4-2 (Sh 1)	-
T10.4-2 (Sh 2)	-
T10.4-2 (Sh 3)	-

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
T10.4-2 (Sh 4)	2
T10.4-2 (Sh 5)	-
T10.4-2 (Sh 5a)	3
T10.4-3	7
F10.4-1	7
F10.4-2	7
F10.4-3	2
F10.4-4 (Sh 1)	7
F10.4-4 (Sh 2)	7
F10.4-5	7
F10.4-6	5
F10.4-7 (Sh 1)	7
F10.4-7 (Sh 2)	7
F10.4-7 (Sh 3)	5
F10.4-7 (Sh 4)	7
F10.4-7 (Sh 5)	6
F10.4-8	5
F10.4-9 (Sh 1)	7
F10.4-9 (Sh 2)	7
F10.4-10	3
F10.4-11	6
F10.4-12 (Sh 1)	3
F10.4-12 (Sh 2)	2
F10.4-12 (Sh 3)	2
F10.4-13 (Sh 1)	7
F10.4-13 (Sh 2)	6
F10.4-14	6
F10.4-15	-
F10.4-16	7
F10.4-17 (Sh 1)	7
F10.4-17 (Sh 2)	7

Chapter 11

Page No.	Rev. No.
11.0-1	6
11.0-2	2
11.0-3	4
11.0-4	-
11.0-5	-
11.0-6	-
11.0-7	-
11.0-8	-
11.1-1	2
11.1-2	2
11.1-2a	2
11.1-3	1
11.1-4	1
11.1-5	-
11.1-6	-
11.1-7	7
11.1-7a	4
11.1-8	-
11.1-9	-
T11.1-1 (Sh 1)	-
T11.1-1 (Sh 2)	-
T11.1-2	-
T11.1-3 (Sh 1)	-
T11.1-3 (Sh 2)	7
T11.1-4 (Sh 1)	-
T11.1-4 (Sh 2)	-
T11.1-4 (Sh 3)	-
T11.1-5	-
11.2-1	-
11.2-2	6
11.2-3	6
11.2-4	7
11.2-5	7
11.2-6	7
11.2-7	7
11.2-7a	7
11.2-8	7
11.2-9	2
11.2-10	7
11.2-11	2
11.2-12	4
T11.2-1 (Sh 1)	-
T11.2-1 (Sh 2)	-
T11.2-1 (Sh 3)	-
T11.2-1 (Sh 4)	-
T11.2-2	-
T11.2-3	-
T11.2-4	7
T11.2-5	-
T11.2-6	-
T11.2-7	-
T11.2-8	-
T11.2-9	-
T11.2-10	-
T11.2-11	-
F11.2-1	4
F11.2-2	4
F11.2-3	2
F11.2-4	6
F11.2-5	-
11.3-1	6
11.3-2	6
11.3-3	-
11.3-4	6
11.3-5	6
11.3-6	-
11.3-7	-
11.3-8	-
11.3-9	-
11.3-10	2
11.3-11	-
11.3-12	-
T11.3-1 (Sh 1)	-
T11.3-1 (Sh 2)	-
T11.3-2 (Sh 1)	-
T11.3-2 (Sh 2)	-
T11.3-3	-
T11.3-4	-
T11.3-5	-
T11.3-6	-
T11.3-7	-
T11.3-8	-
T11.3-9	-
F11.3-1	3
F11.3-2 (Sh 1)	6
F11.3-2 (Sh 2)	6
F11.3-2 (Sh 3)	7
F11.3-2 (Sh 4)	4
F11.3-3	6
F11.3-4	7
F11.3-5	-
11.4-1	4
11.4-2	4
11.4-3	4
11.4-4	6
11.4-4a	6
11.4-5	4
11.4-6	4
11.4-7	4
11.4-8	4
11.4-9	4
11.4-10	4
11.4-11	4
11.4-12	4
11.4-13	4
11.4-14	4
11.4-15	4
11.4-16	4
11.4-17	5
11.4-18	5
11.4-19	4
11.4-20	4
T11.4-1	-
T11.4-2	-
T11.4-3	-
T11.4-4	-
T11.4-5	-
T11.4-6	-
T11.4-7	-
T11.4-8 (Sh 1)	-
T11.4-8 (Sh 2)	-
T11.4-8 (Sh 3)	-
T11.4-8 (Sh 4)	-
T11.4-8 (Sh 5)	-
T11.4-8 (Sh 6)	-
T11.4-8 (Sh 7)	-
T11.4-9	-
T11.4-10	-
F11.4-1 (Sh 1)	4
F11.4-1 (Sh 2)	-
F11.4-2	4
F11.4-3	2
F11.4-4	-
F11.4-5	-
F11.4-6	-
F11.4-7	-
F11.4-8	4
F11.4-9	2
F11.4-10	5
F11.4-11 (Sh 1)	6
F11.4-11 (Sh 2)	6
11.5-1	-
11.5-2	6
11.5-3	-
11.5-4	7
11.5.4a	7
11.5-5	3
11.5-6	7
11.5-7	6
11.5-8	7
11.5-9	4
11.5-10	4
11.5-11	7
11.5-12	4
T11.5-1 (Sh 1)	7
T11.5-1 (Sh 2)	2
T11.5-1 (Sh 3)	7
T11.5-2 (Sh 1)	6
T11.5-2 (Sh 2)	-
T11.5-3 (Sh 1)	-
T11.5-3 (Sh 2)	-
T11.5-3 (Sh 3)	-
F11.5-1	4

Appendix 11A

Page No.	Rev. No.
Cover Sheet	-
11A-1	
11A-2	-
11A-3	-
11A-4	7
11A-4a	4
11A-5	-
11A-6	-
11A-7	-
11A-8	-
11A-9	-
11A-10	-
T11A-1	-
T11A-2	-
T11A-3 (Sh 1)	-
T11A-3 (Sh 2)	-
T11A-4	-
F11A-1	-
F11A-2	7
F11A-3	-
F11A-4	-

VOLUME 11

Chapter 12

Page No.	Rev. No.
12.0-1	6
12.0-2	-
12.0-3	6
12.0-4	4
12.0-5	-
12.0-6	-
12.0-7	-
12.0-8	-
12.0-9	-
12.0-10	4
12.1-1	6
12.1-2	6

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
12.1-3	6
12.1-4	6
12.1-4a	6
12.1-5	-
12.1-6	6
12.1-7	-
12.1-8	-
12.1-9	-
12.1-10	-
12.1-11	6
12.1-12	6
12.1-12a	6
12.1-13	6
12.1-14	-
12.2-1	-
12.2-2	-
12.2-3	5
12.2-3a	5
12.2-4	-
12.2-5	-
12.2-6	-
12.2-7	-
12.2-8	1
12.2-9	-
12.2-10	-
12.2-11	-
12.2-12	-
12.2-13	-
12.2-14	6
12.2-14a	6
12.2-15	-
12.2-16	-
12.2-17	2
12.2-18	2
12.2-19	2
12.2-20	-
T12.2-1	-
T12.2-2	6
T12.2-3	-
T12.2-4	-
T12.2-5 (Sh 1)	-
T12.2-5 (Sh 2)	-
T12.2-5 (Sh 3)	-
T12.2-5 (Sh 4)	-
T12.2-5 (Sh 5)	-
T12.2-6 (Sh 1)	-
T12-2-6 (Sh 2)	-
T12.2-6 (Sh 3)	-
T12.2-7	-
T12.2-8	-
T12.2-9 (Sh 1)	-
T12.2-9 (Sh 2)	-
T12.2-9 (Sh 3)	-
T12.2-10	-
T12.2-11 (Sh 1)	-
T12.2-11 (Sh 2)	-
T12.2-11 (Sh 3)	-
T12.2-12 (Sh 1)	-
T12.2-12 (Sh 2)	-
T12.2-13	-
T12.2-14	-
T12.2-15	-
T12.2-16	-
T12.2-17	-
T12.2-18	-
T12.2-19 (Sh 1)	-
T12.2-19 (Sh 2)	-
T12.2-20	-
T12.2-21	-
T12.2-22	-
T12.2-23	-
T12.2-24	-
T12.2-25 (Sh 1)	-
T12.2-25 (Sh 2)	-
T12.2-26	-
T12.2-27 (Sh 1)	-
T12.2-27 (Sh 2)	-
T12.2-28	-
T12.2-29 (Sh 1)	-
T12.2-29 (Sh 2)	-
T12.2-30 (Sh 1)	-
T12.2-30 (Sh 2)	-
T12.2-30 (Sh 3)	-
T12.2-31	-
T12.2-32	-
T12.2-33	-
T12.2-34	-
T12.2-35	-
T12.2-36	-
T12.2-37	-
12.3-1	-
12.3-2	2
12.3-3	-
12.3-4	6
12.3-5	6
12.3-5a	6
12.3-6	4
12.3-6a	4
12.3-7	1
12.3-8	6
12.3-9	1
12.3-10	4
12.3-10a	4
12.3-11	5
12.3-12	1
12.3-13	2
12.3-14	2
12.3-15	2
12.3-16	2
12.3-17	2
12.3-18	2
12.3-19	7
12.3-20	2
12.3-20a	2
12.3-21	1
12.3-22	6
12.3-23	1
12.3-24	6
12.3-25	4
12.3-26	2
12.3-27	2
T12.3-1	-
T12.3-2	-
T12.3-3	-
T12.3-4	-
T12.3-5	-
T12.3-6	-
T12.3-7	-
T12.3-8	-
T12.3-9	-
T12.3-10 (Sh 1)	-
T12.3-10 (Sh 2)	-
T12.3-11	-
T12.3-12	-
T12.3-13 (Sh 1)	-
T12.3-13 (Sh 2)	-
T12.3-14 (Sh 1)	7
T12.3-14 (Sh 2)	-
T12.3-15 (Sh 1)	4
T12.3-15 (Sh 1a)	6
T12.3-15 (Sh 2)	-
T12.3-16	2
F12.3-1	-
F12.3-2	-
F12.3-3	4
F12.3-4	-
F12.3-5	-
F12.3-6	-
F12.3-7	-
F12.3-8	-
F12.3-9	6
F12.3-10	1
F12.3-11	-
F12.3-12	1
F12.3-13	-
F12.3-14	-
F12.3-15	2
F12.3-16	2
F12.3-17	4
F12.3-18	-
F12.3-19 (Sh 1)	5
F12.3-19 (Sh 2)	-
F12.3-20 (Sh 1)	5
F12.3-20 (Sh 2)	4
12.4-1	-
12.4-2	6
12.4-3	6
12.4-4	6
12.4-4a	6
12.4-5	-
T12.4-1	6
T12.4-2 (Sh 1)	-
T12.4-2 (Sh 2)	-
T12.4-3 (Sh 1)	-
T12.4-3 (Sh 2)	-
T12.4-4 (Sh 1)	6
T12.4-4 (Sh 2)	4
T12.4-4 (Sh 3)	6
T12.4-5	-
T12.4-6 (Sh 1)	-
T12.4-6 (Sh 2)	-
T12.4-7	-
12.5-1	6
12.5-1a	6
12.5-2	6
12.5-3	6
12.5-4	6
12.5-5	6
12.5-6	4
12.5-7	6
12.5-8	6
12.5-9	7
12.5-10	7
12.5-11	4
12.5-12	4
T12.5-1	4
T12.5-2	6
F12.5-1	4
Chapter 13	
13.0-1	7
13.0-2	6
13.0-3	2
13.0-4	4
13.0-5	2
13.1-1	7
13.1-2	7
13.1-3	7
13.1-4	6
13.1-5	7
13.1-6	6
13.1-7	7
13.1-8	7
13.1-9	7
13.1-10	7
13.1-11	7
13.1-12	7
13.1-13	7
13.1-14	7
13.1-15	7
F13.1-1	6
F13.1-2	7
F13.1-3	6
F13.1-4	7
13.2-1	6
13.2-2	5
13.2-3	6
13.2-4	5
13.2-5	6
13.2-5a	6
13.2-6	5
13.2-7	5
13.2-8	7
13.2-8a	7
13.2-9	6
13.2-10	5
13.2-11	5
13.2-12	5
13.2-13	5

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
13.3-1	6
13.4-1	5
13.4-2	7
13.4-3	7
13.4-4	4
13.4-5	4
13.5-1	4
13.5-1a	4
13.5-2	-
13.5-3	5
13.5-4	-
13.5-5	-
13.5-6	-
13.5-7	-
13.5-8	-
13.5-9	-
13.5-10	-
13.5-11	-
13.5-12	-
13.5-13	-
13.5-14	-
13.5-15	-
13.5-16	-
13.5-17	7
13.5-18	-
13.5-19	-
13.5-20	-
13.5-21	4
13.5-22	-
13.5-23	-
13.5-24	-
13.5-25	-
13.5-26	-
13.5-27	7
13.5-28	-
13.5-29	6
13.5-30	6
13.5-31	2
13.5-32	5
T13.5-1	4
T13.5-2 (Sh 1)	-
T13.5-2 (Sh 2)	-
T13.5-2 (Sh 3)	-
T13.5-2 (Sh 4)	-
T13.5-2 (Sh 5)	-
F13.5-1	4
13.6-1	-

Appendix 13A

Page No.	Rev. No.
All Sheets	-

Chapter 14

Page No.	Rev. No.
14.0-1	-
14.0-2	-
14.1-1	-
14.2-1	-
14.2-2	-
14.2-3	-
14.2-4	-
14.2-5	6
14.2-6	-
14.2-7	-
14.2-8	-
14.2-9	-
14.2-10	-
14.2-11	-
14.2-12	-
14.2-13	-
14.2-14	-
14.2-15	-
14.2-16	-
T14.2-1 (Sh 1)	-
T14.2-1 (Sh 2)	-
T14.2-1 (Sh 3)	-
T14.2-2	-
T14.2-3 (Sh 1)	-
T14.2-3 (Sh 2)	-
T14.2-3 (Sh 3)	-
T14.2-3 (Sh 4)	-
T14.2-3 (Sh 5)	-
T14.2-3 (Sh 6)	-
T14.2-3 (Sh 7)	-
T14.2-3 (Sh 8)	-
T14.2-3 (Sh 9)	-
T14.2-3 (Sh 10)	-
T14.2-3 (Sh 11)	-
T14.2-3 (Sh 12)	-
T14.2-3 (Sh 13)	-
T14.2-3 (Sh 14)	-
T14.2-3 (Sh 15)	-
T14.2-3 (Sh 16)	-
T14.2-3 (Sh 17)	-
T14.2-3 (Sh 18)	5
T14.2-3 (Sh 19)	-
T14.2-3 (Sh 20)	-
T14.2-3 (Sh 21)	-
T14.2-3 (Sh 22)	-
T14.2-3 (Sh 23)	-
T14.2-3 (Sh 24)	-
T14.2-3 (Sh 25)	-
T14.2-3 (Sh 26)	-
T14.2-3 (Sh 27)	-
T14.2-3 (Sh 28)	-
T14.2-3 (Sh 29)	-
T14.2-3 (Sh 30)	-
T14.2-3 (Sh 31)	-
T14.2-3 (Sh 32)	-
T14.2-3 (Sh 33)	-
T14.2-3 (Sh 34)	-
T14.2-3 (Sh 35)	-
T14.2-3 (Sh 36)	-
T14.2-3 (Sh 37)	-
T14.2-3 (Sh 38)	-
T14.2-3 (Sh 39)	-
T14.2-3 (Sh 40)	-
T14.2-3 (Sh 41)	-
T14.2-3 (Sh 42)	-
T14.2-3 (Sh 43)	-
T14.2-3 (Sh 44)	-
T14.2-3 (Sh 45)	-
T14.2-3 (Sh 46)	-
T14.2-3 (Sh 47)	-
T14.2-3 (Sh 48)	-
T14.2-3 (Sh 49)	-
T14.2-4 (Sh 1)	-
T14.2-4 (Sh 2)	-
T14.2-4 (Sh 3)	-
T14.2-4 (Sh 4)	-
T14.2-4 (Sh 5)	-
T14.2-4 (Sh 6)	-
T14.2-4 (Sh 7)	-
T14.2-4 (Sh 8)	-
T14.2-4 (Sh 9)	-
T14.2-4 (Sh 10)	-
T14.2-4 (Sh 11)	-
T14.2-4 (Sh 12)	-
T14.2-4 (Sh 13)	-
T14.2-4 (Sh 14)	-
T14.2-4 (Sh 15)	-
T14.2-4 (Sh 16)	-
T14.2-4 (Sh 17)	-
T14.2-4 (Sh 18)	-
T14.2-4 (Sh 19)	-
T14.2-4 (Sh 20)	-
T14.2-4 (Sh 21)	-
T14.2-4 (Sh 22)	-
T14.2-4 (Sh 23)	-
T14.2-4 (Sh 24)	-
T14.2-4 (Sh 25)	-
T14.2-4 (Sh 26)	-
T14.2-4 (Sh 27)	-
T14.2-4 (Sh 28)	-
T14.2-4 (Sh 29)	-
T14.2-4 (Sh 30)	-
T14.2-4 (Sh 31)	-
T14.2-4 (Sh 32)	-
T14.2-4 (Sh 33)	-
T14.2-4 (Sh 34)	-
T14.2-4 (Sh 35)	-
T14.2-5 (Sh 1)	-
T14.2-5 (Sh 2)	-
T14.2-5 (Sh 3)	-
T14.2-5 (Sh 4)	-
T14.2-5 (Sh 5)	-
T14.2-5 (Sh 6)	-
T14.2-5 (Sh 7)	-
T14.2-5 (Sh 8)	-
T14.2-5 (Sh 9)	-
T14.2-5 (Sh 10)	-
T14.2-5 (Sh 11)	-
T14.2-5 (Sh 12)	-
T14.2-5 (Sh 13)	-
T14.2-5 (Sh 14)	-
T14.2-5 (Sh 15)	-
T14.2-5 (Sh 16)	-
T14.2-5 (Sh 17)	-
T14.2-5 (Sh 18)	-
T14.2-5 (Sh 19)	-
T14.2-5 (Sh 20)	-
T14.2-5 (Sh 21)	-
T14.2-5 (Sh 22)	-
T14.2-5 (Sh 23)	-
T14.2-5 (Sh 24)	-
T14.2-5 (Sh 25)	-
T14.2-5 (Sh 26)	-
T14.2-5 (Sh 27)	-
T14.2-5 (Sh 28)	-
T14.2-5 (Sh 29)	-
T14.2-5 (Sh 30)	-
T14.2-5 (Sh 31)	-
T14.2-5 (Sh 32)	-
T14.2-5 (Sh 33)	-
T14.2-5 (Sh 34)	-
T14.2-5 (Sh 35)	-
T14.2-5 (Sh 36)	-
T14.2-5 (Sh 37)	-
T14.2-5 (Sh 38)	-
T14.2-5 (Sh 39)	-
T14.2-5 (Sh 40)	-
T14.2-5 (Sh 41)	-
T14.2-5 (Sh 42)	-
T14.2-5 (Sh 43)	-
T14.2-5 (Sh 44)	-
T14.2-5 (Sh 45)	-
T14.2-5 (Sh 46)	-
T14.2-5 (Sh 47)	-
T14.2-5 (Sh 48)	-
T14.2-5 (Sh 49)	-

VOLUME 12

Chapter 15

Page No.	Rev. No.
15.0-i	7
15.0-ii	7
15.0-iii	7
15.0-iv	7
15.0-v	7
15.0-vi	7
15.0-vii	7
15.0-viii	7
15.0-ix	7
15.0-x	7
15.0-xi	7
15.0-xii	7
15.0-xiii	7
15.0-xiv	7
15.0-xv	7
15.0-xvi	7
15.0-xvii	7
15.0-xviii	7
15.0-xix	7
15.0-xx	7
15.0-xxi	7
15.0-xxii	7
15.0-xxiii	7
15.0-xxiv	7
15.0-xxv	7

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
15.0-1	4
15.0-2	4
15.0-3	4
15.0-4	4
15.0-5	4
15.0-6	6
15.0-7	7
15.0-7a	7
15.0-7b	7
15.0-8	6
15.0-9	4
15.0-10	4
15.0-11	4
15.0-12	7
15.0-13	7
15.0-14	7
15.0-14a	7
15.0-15	7
15.0-16	7
T15.0-1	4
T15.0-2	4
T15.0-3 (Sh 1)	7
T15.0-3 (Sh 2)	7
T15.0-3 (Sh 3)	7
T15.0-4 (Sh 1)	6
T15.0-4 (Sh 2)	6
T15.0-5 (Sh 1)	6
T15.0-5 (Sh 2)	6
T15.0-5 (Sh 3)	6
T15.0-6	-
T15.0-7 (Sh 1)	6
T15.0-7 (Sh 2)	6
T15.0-8	-
F15.0-1	7
F15.0-2	7
F15.0-3	4
F15.0-4	7
F15.0-5	7
F15.0-6	4
F15.0-7	-
F15.0-8	6
F15.0-9	6
F15.0-10	6
F15.0-11	6
F15.0-12	6
F15.0-13	6
F15.0-14	6
F15.0-15	6
F15.0-16	6
F15.0-17	6
F15.0-18	6
F15.0-20	6
F15.0-21	6
F15.0-22	6
F15.0-23	6
F15.0-24	6
F15.0-25	6
F15.0-26	-
F15.0-27	-
F15.0-28	-
F15.0-29	-
F15.0-30	-
F15.0-31	-
F15.0-32	4
15.1-1	4
15.1-2	4
15.1-3	4
15.1-4	7
15.1-5	7
15.1-6	7
15.1-7	7
15.1-8	7
15.1-9	7
15.1-10	7
15.1-11	7
15.1-12	7
15.1-13	7
15.1-14	7
15.1-15	7
15.1-16	7
15.1-17	7
15.1-18	7
15.1-19	7
15.1-20	7
T15.1-1 (Sh 1)	7
T15.1-1 (Sh 2)	7
T15.1-2 (Sh 1)	-
T15.1-2 (Sh 2)	-
T15.1-3	-
T15.1-4	-
T15.1-5	-
T15.1-6	-
F15.1-1 (Sh 1)	7
F15.1-1 (Sh 2)	7
F15.1-1 (Sh 3)	7
F15.1-1 (Sh 4)	7
F15.1-2 (Sh 1)	7
F15.1-2 (Sh 2)	7
F15.1-3 (Sh 1)	7
F15.1-3 (Sh 2)	7
F15.1-4 (Sh 1)	7
F15.1-4 (Sh 2)	7
F15.1-5 (Sh 1)	7
F15.1-5 (Sh 2)	7
F15.1-6 (Sh 1)	7
F15.1-6 (Sh 2)	7
F15.1-6 (Sh 3)	7
F15.1-6 (Sh 4)	7
F15.1-6 (Sh 5)	7
F15.1-6 (Sh 6)	7
15.2-1	4
15.2-2	4
15.2-3	4
15.2-4	7
15.2-5	7
15.2-6	7
15.2-7	7
15.2-8	7
15.2-9	7
15.2-10	7
15.2-11	7
15.2-12	7
15.2-13	7
15.2-14	7
15.2-15	7
15.2-16	7
15.2-17	7
15.2-18	7
15.2-19	7
15.2-20	7
15.2-21	7
15.2-22	5
15.2-23	5
T15.2-1 (Sh 1)	7
T15.2-1 (Sh 2)	7
T15.2-1 (Sh 3)	7
T15.2-2	7
F15.2-1 (Sh 1)	7
F15.2-1 (Sh 2)	7
F15.2-1 (Sh 3)	7
F15.2-1 (Sh 4)	7
F15.2-2	7
F15.2-3	7
F15.2-4	7
F15.2-5 (Sh 1)	7
F15.2-5 (Sh 2)	7
F15.2-5 (Sh 3)	7
F15.2-5 (Sh 4)	7
F15.2-6 (Sh 1)	7
F15.2-6 (Sh 2)	7
F15.2-6 (Sh 3)	7
F15.2-6 (Sh 4)	7
F15.2-7 (Sh 1)	7
F15.2-7 (Sh 2)	7
F15.2-7 (Sh 3)	7
F15.2-7 (Sh 4)	7
F15.2-7 (Sh 5)	7
15.3-1	7
15.3-2	7
15.3-3	7
15.3-4	7
15.3-5	7
15.3-6	7
15.3-7	7
15.3-8	7
15.3-9	7
15.3-10	7
15.3-11	7
15.3-12	7
15.3-13	7
15.3-14	7
T15.3-1	7
T15.3-2	7
T15.3-3 (Sh 1)	-
T15.3-3 (Sh 2)	-
T15.3-4	-
T15.3-5	-
T15.3-6	-
T15.3-7	-
T15.3-8	-
T15.3-9	7
F15.3-1 (Sh 1)	7
F15.3-1 (Sh 2)	7
F15.3-1 (Sh 3)	7
F15.3-2	7
F15.3-3	7
F15.3-4	7
F15.3-5	7
F15.3-6 (Sh 1)	7
F15.3-6 (Sh 2)	7
F15.3-6 (Sh 3)	7
F15.3-6 (Sh 4)	7
F15.3-7	7
15.4-1	7
15.4-2	7
15.4-3	7
15.4-4	7
15.4-5	7
15.4-6	7
15.4-7	7
15.4-8	7
15.4-9	7
15.4-10	7
15.4-11	7
15.4-12	7
15.4-13	7
15.4-14	7
15.4-15	7
15.4-16	7
15.4-17	7
15.4-18	7
15.4-19	7
15.4-20	7
15.4-21	7
15.4-22	7
15.4-23	7
15.4-24	7
15.4-25	7
15.4-26	7
15.4-27	7
15.4-28	7
15.4-29	7
15.4-30	7
15.4-31	7
15.4-32	7
15.4-33	7
15.4-34	7
15.4-35	7
15.4-36	7
15.4-37	7
15.4-38	7
15.4-39	7
15.4-40	7
15.4-41	7
15.4-42	7
T15.4-1 (Sh 1)	7
T15.4-1 (Sh 2)	7
T15.4-1 (Sh 3)	7
T15.4-1 (Sh 4)	7
T15.4-1 (Sh 5)	7
T15.4-2	7

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
T15.4-3 (Sh 1)	-
T15.4-3 (Sh 2)	-
T15.4-4	-
T15.4-5	3
T15.4-6	3
T15.4-7	-
T15.4-8	-
T15.4-9	-
T15.4-10	-
T15.4-11	-
T15.4-12	3
F15.4-1	7
F15.4-2	7
F15.4-3	7
F15.4-4 (Sh 1)	7
F15.4-4 (Sh 2)	7
F15.4-4 (Sh 3)	7
F15.4-5 (Sh 1)	7
F15.4-5 (Sh 2)	7
F15.4-5 (Sh 3)	7
F15.4-6	7
F15.4-7	7
F15.4-8	7
F15.4-9 (Sh 1)	7
F15.4-9 (Sh 2)	7
F15.4-10 (Sh 1)	7
F15.4-10 (Sh 2)	7
F15.4-11 (Sh 1)	7
F15.4-11 (Sh 2)	7
15.5-1	7
15.5-2	7
15.5-3	7
15.5-4	7
15.5-5	7
15.5-6	7
15.5-7	7
15.5-8	7
T15.5-1	7
T15.5-2	7
T15.5-3	7
T15.5-4	7
T15.5-5	7
F15.5-1 (Sh 1)	7
F15.5-1 (Sh 2)	7
F15.5-1 (Sh 3)	7
F15.5-2	7
F15.5-3	7
F15.5-4	7
F15.5-5	7
15.6-1	7
15.6-2	7
15.6-3	7
15.6-4	4
15.6-5	4
15.6-6	4
15.6-7	4
15.6-8	4
15.6-9	4
15.6-10	4
15.6-11	4
15.6-12	4
15.6-13	4
15.6-14	7
15.6-14a	7
15.6-14b	7
15.6-15	4
15.6-16	6
15.6-17	7
15.6-17a	7
15.6-17b	7
15.6-17c	7
15.6-18	4
15.6-19	6
15.6-20	4
15.6-21	4
15.6-22	4
15.6-23	4
15.6-24	4
15.6-25	4
15.6-26	4
15.6-27	7
15.6-28	4
15.6-29	4
15.6-30	4
15.6-31	4
15.6-32	4
15.6-33	4
15.6-34	4
15.6-35	4
15.6-36	6
15.6-37	4
15.6-38	7
T15.6-1 (Sh 1)	7
T15.6-1 (Sh 2)	4
T15.6-1 (Sh 3)	4
T15.6-1 (Sh 4)	4
T15.6-2 (Sh 1)	-
T15.6-2 (Sh 2)	-
T15.6-3	-
T15.6-4	-
T15.6-5	-
T15.6-6 (Sh 1)	6
T15.6-6 (Sh 2)	6
T15.6-7	6
T15.6-8	4
T15.6-9	6
T15.6-10	6
T15.6-11	6
T15.6-12	6
T15.6-13	4
T15.6-14	4
T15.6-15	4
T15.6-16 (Sh 1)	-
T15.6-16 (Sh 2)	-
T15.6-17	-
T15.6-18	-
T15.6-19	4
T15.6-20	-
T15.6-21	-
T15.6-22	-
T15.6-23	4
T15.6-24	-
T15.6-25	-
F15.6-1	7
F15.6-2	7
F15.6-3	4
F15.6-4	4
F15.6-5	4
F15.6-6	4
F15.6-7	4
F15.6-8	4
F15.6-9	4
F15.6-10	4
F15.6-11	4
F15.6-12	4
F15.6-13	4
F15.6-14	4
F15.6-15	4
F15.6-16	4
F15.6-17	4
F15.6-18	4
F15.6-19	4
F15.6-20	4
F15.6-21	4
F15.6-22	4
F15.6-23	4
F15.6-24	4
F15.6-25	4
F15.6-26	4
F15.6-27	4
F15.6-28	4
F15.6-29	4
F15.6-30	4
F15.6-31	4
F15.6-32	4
F15.6-33	4
F15.6-34	4
F15.6-35	4
F15.6-36	4
F15.6-37	4
F15.6-38	4
F15.6-39	4
F15.6-40	4
F15.6-41	4
F15.6-42	4
F15.6-43	4
F15.6-44	4
F15.6-45	4
F15.6-46	4
F15.6-47	4
F15.6-48	4
F15.6-49	4
F15.6-50	6
F15.6-51	6
F15.6-52	6
F15.6-53	6
F15.6-54	6
F15.6-55	6
F15.6-56	6
F15.6-57	6
F15.6-58	6
F15.6-59	6
F15.6-60	6
15.7-1	-
15.7-2	2
15.7-3	-
15.7-4	-
15.7-5	-
15.7-6	-
15.7-7	-
15.7-8	-
15.7-9	-
15.7-10	7
15.7-10a	7
15.7-11	4
15.7-12	7
15.7-13	7
15.7-14	7
15.7-15	7
15.7-16	7
15.7-17	7
15.7-17a	7
15.7-17b	7
15.7-18	4
T15.7-1	-
T15.7-2 (Sh 1)	-
T15.7-2 (Sh 2)	-
T15.7-3	-
T15.7-4	-
T15.7-5	-
T15.7-6	2
T15.7-7 (Sh 1)	-
T15.7-7 (Sh 2)	-
T15.7-8	-
T15.7-9	-
T15.7-10	-
T15.7-11	-
T15.7-12	-
T15.7-13	-
T15.7-14	-
T15.7-15	-
T15.7-16	-
T15.7-17	-
T15.7-18 (Sh 1)	7
T15.7-18 (Sh 2)	6
T15.7-18 (Sh 3)	4
T15.7-19	6
T15.7-20	-
T15.7-21	7
T15.7-22	-
T15.7-23	7
T15.7-24	7
T15.7-24a	4
T15.7-25	7
T15.7-25a	7
T15.7-26 (Sh 1)	-
T15.7-26 (Sh 2)	-
T15.7-27	-
T15.7-28	-
15.8-1	4
15.9-1	3

LIST OF EFFECTIVE PAGES

Page No.	Rev. No.
Appendix 15A	
All Sheets	-
Appendix 15B	
15B-1	6
15B-1a	6
15B-2	-
15B-3	-
15B-4	4
15B-5	-
15B-6	-
15B-7	1
15B-8	1
15B-9	1
15B-10	1
15B-11	1
T15B-1	-
T15B-2	-
T15B-3	-
T15B-4	-
T15B-5	-
T15B-6	1
VOLUME 13	
Chapter 16	
16.0-1	-
16.1-1	6
Chapter 17	
17.0-1	6
17.0-2	6
17.0-3	6
17.0-4	6
17.0-5	7
17.0-6	-
17.0-7	-
17.1-1	-
17.1-2	-
17.1-3	-
17.1-4	-
17.1-5	-
17.1-6	-
17.1-7	-
17.1-8	-
17.1-9	-
17.1-10	-
17.1-11	-
17.1-12	-
17.1-13	-
17.1-14	-
17.1-15	-
17.1-16	-
17.1-17	-
17.1-18	-
17.1-19	-
17.1-20	-
17.1-21	-
17.1-22	-
17.1-23	-
17.1-24	-
17.1-25	-
17.1-26	-
17.1-27	-
17.1-28	-
17.1-29	-
17.1-30	-
17.1-31	-
17.1-32	-
17.1-33	-
17.1-34	-
17.1-35	-
17.1-36	-
17.1-37	-
17.1-38	-
17.1-39	-

Page No.	Rev. No.
17.1-40	-
T17.1-1	-
T17.1-2 (Sh 1)	-
T17.1-2 (Sh 2)	-
T17.1-3 (Sh 1)	-
T17.1-3 (Sh 2)	-
T17.1-3 (Sh 3)	-
T17.1-3 (Sh 4)	-
T17.1-3 (Sh 5)	-
T17.1-3 (Sh 6)	-
T17.1-4 (Sh 1)	-
T17.1-4 (Sh 2)	-
T17.1-5	-
F17.1-1	-
F17.1-2	-
17.2-1	6
17.2-2	6
17.2-3	7
17.2-4	6
17.2-5	7
17.2-6	7
17.2-7	7
17.2-8	7
17.2-9	7
17.2-10	7
17.2-11	6
17.2-12	6
17.2-13	6
17.2-14	7
17.2-15	6
17.2-16	6
17.2-17	6
17.2-18	6
17.2-19	6
17.2-20	6
17.2-21	6
17.2-22	6
17.2-23	6
17.2-24	6
17.2-25	6
17.2-26	6
17.2-27	6
17.2-28	6
17.2-29	6
17.2-30	6
17.2-31	6
17.2-32	6
17.2-33	6
17.2-34	6
17.2-35	6
17.2-36	6
17.2-37	7
17.2-38	6
17.2-39	7
17.2-40	7
F17.2-1	7
Appendix 17A	
Cover Sheet	4
17A-1	5
17A-2	6
17A-2a	5
17A-3	7
17A-4	7
17A-4a	7
17A-5	3
17A-6	6
17A-7	5
17A-8	2
17A-9	2
17A-10	2
17A-11	2
17A-12	2
17A-13	2
17A-14	2
17A-15	2
17A-16	7
17A-17	6

Page No.	Rev. No.
Appendix 17B	
Cover Sheet	6
Appendix 17C	
Cover Sheet	7
17C-1	7
17C-2	7
17C-3	7
17C-4	7
List of Effective Pages	
1 - 25	7

Amendment History

Letters to NRC and associated updating instructions for the FSAR.